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TECHNICAL PROGRESS REVIEW

POWER REACTOR TECHNOLOGY

PREPARED BY GENERAL NUCLEAR ENGINEERING CORP. FOR THE U S ATOMIC ENERGY COMMISSION



DECEMBER 1957

VOLUME 1

NUMBER 1

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Erratum

("Power Reactor Technology" journal, December 1957, Vol. I, No. 1, p. 24)

Expressions for parasitic absorption cross sections of the additives to uranium (zirconium, niobium, molybdenum and silicon) should be multiplied by the factor 0.01.



POWER REACTOR TECHNOLOGY

a review of recent developments prepared by

GENERAL NUCLEAR ENGINEERING CORP.

DECEMBER 1957

VOLUME I

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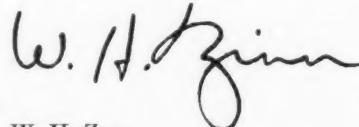


foreword

This quarterly review of reactor development has been prepared at the request of the Division of Information Services of the United States Atomic Energy Commission. Its purpose is to assist interested organizations in the task of keeping abreast of new results in reactor technology for civilian application.

The report is a concise discussion of selected phases of research and development for which there have been significant advances or a heightened interest in the past few months. It is not meant to be a comprehensive abstract of all material published during the quarter,* nor is it meant to be a treatise on any part of the subject.

In the overall plan of the report, it is intended that various subjects will be treated from time to time and will be brought up to date at that time. Any interpretation of results which is given represents only the opinion of the editors of the report, who are General Nuclear Engineering Corporation personnel. Readers are urged to consult the original references wherever possible in order to obtain all the background of the work reported and to obtain the interpretation of the results given by the original authors.



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Dunedin, Fla.

*This issue of *Power Reactor Technology* summarizes developments in the civilian power reactor field for the first quarter of 1957; subsequent issues of the journal will review reactor development progress on a current basis.

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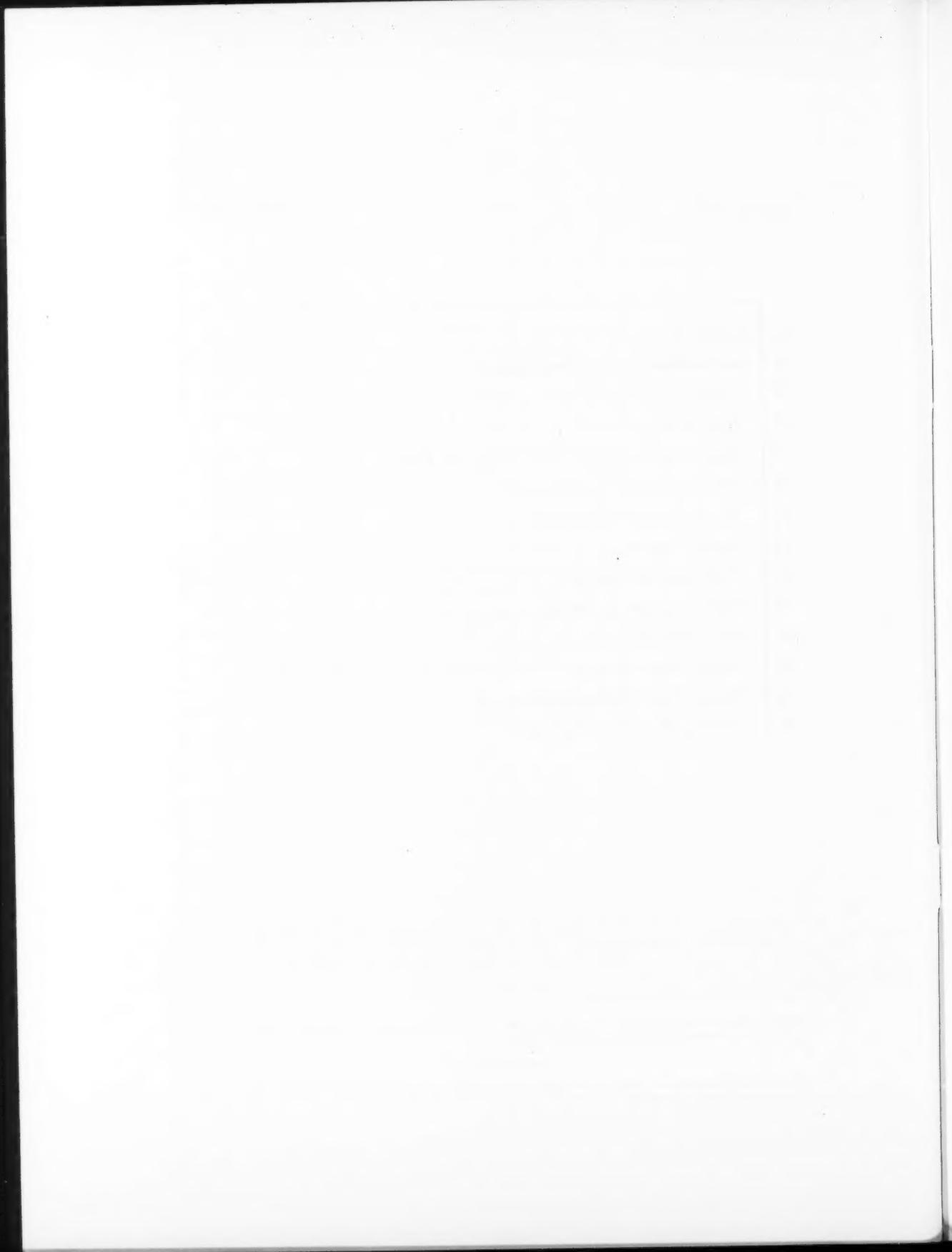
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GENERAL RESEARCH AND DEVELOPMENT

APPLICATIONS, PERFORMANCE, AND EVALUATIONS

Fuel Prices

The recently announced AEC prices¹ for uranium, in the form of UF_6 , are plotted as a function of enrichment in figure 1. Also plotted, as a smoothed curve, is the cost per gram of U^{235} in excess of that contained in natural uranium. As had been anticipated,

that cost is relatively high, even for very slight enrichment. The conclusion may be drawn that the cost of fuel for slightly enriched reactors (of, say, 2 percent enrichment and above) will be high unless either reprocessing costs are low or very long burnup is achieved. Figure 2 shows the fuel lifetime necessary to achieve a fuel burnup cost as low as 1 mill per

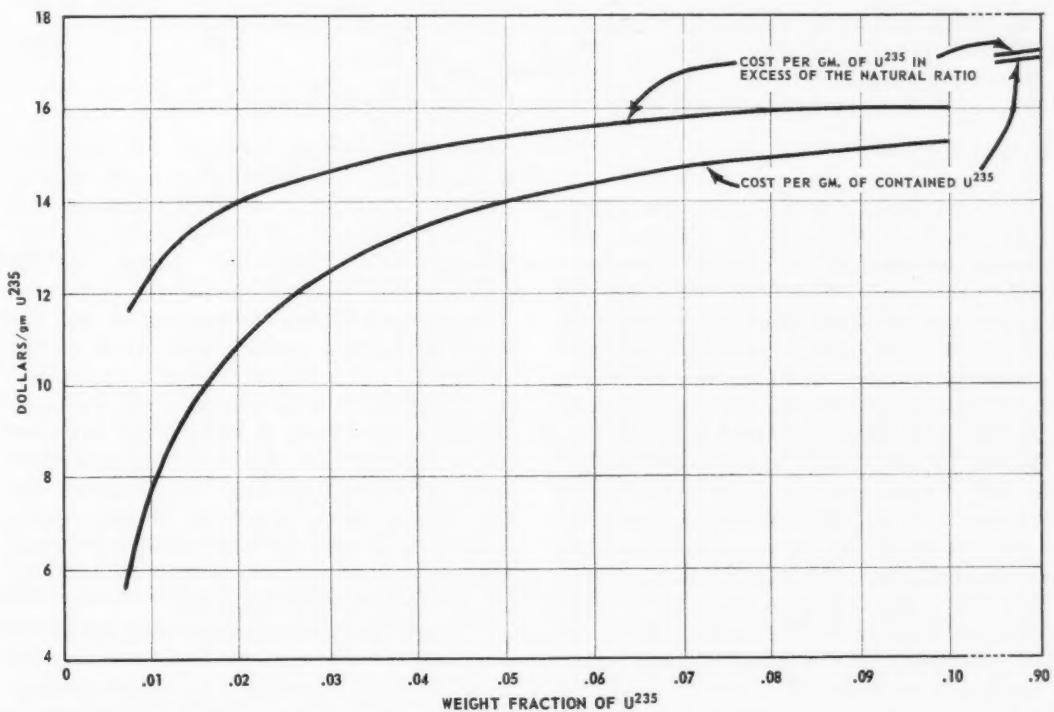


Figure 1. AEC fuel prices (Significance of upper curve is as follows. Fuel containing any given weight fraction (f) of U^{235} can be considered to contain a weight fraction 0.00714 of U^{235} originally present in the natural uranium plus a weight fraction ($f-0.00714$) of U^{235} "added" by the enrichment process. Upper curve shows cost of this "added" weight fraction in dollars per gram of U^{235} "added.")

kilowatt-hour with "throwaway" fuel, as a function of enrichment. A plant efficiency of 25 percent is assumed.

price index. The term of the contracts will be from the respective dates of execution until June 30, 1967, unless the required services

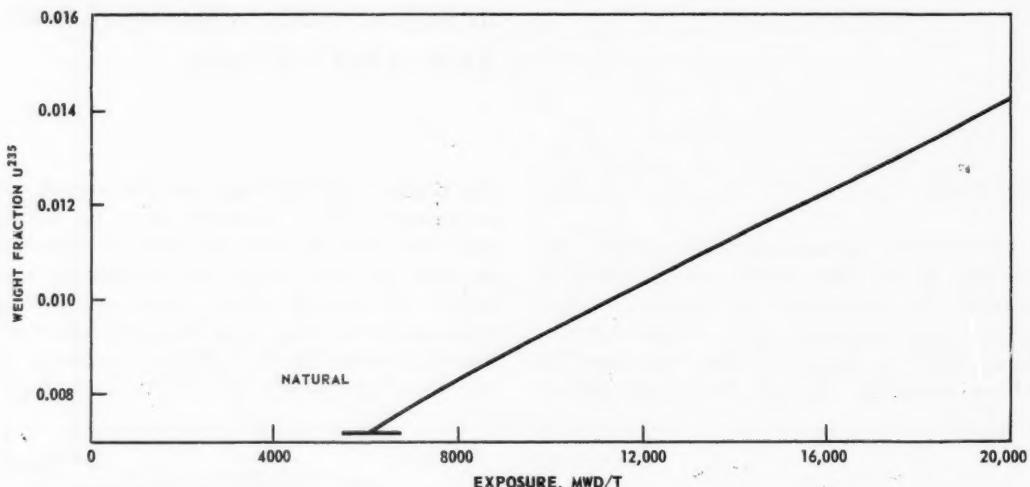


Figure 2. Fuel lifetime necessary to achieve a fuel burnup cost as low as 1 mill/kw-hr with no reprocessing.

Reprocessing Costs

It is quite evident from figure 2 that attractively low fuel cost cannot be obtained with "throwaway" fuel of enrichment appreciably higher than the natural ratio. The hope for low fuel cost with the higher enrichments rests on the possibility of reprocessing the fuel after it is discharged from the reactor and recovering the unburned U²³⁵ as well as any fissionable product which has been formed by conversion. The net fuel cost will therefore depend upon the cost of reprocessing, and low fuel costs can be obtained only if the reprocessing cost is low.

The Atomic Energy Commission has recently made a preliminary statement of the basis upon which the charges for reprocessing by Commission-operated plants will be established over the next 10 years.² The announcement states that the Commission will undertake to provide fuel processing for persons licensed under Sections 103 or 104 of the Atomic Energy Act, on the basis of individual contracts which set a processing charge. The charge is firm except for escalation clauses tied to some appropriate

become commercially available at reasonable prices before that time. The charge will be determined separately for each contract and may vary to some extent, depending upon the difficulty of the "head end" process and the necessity for conversion of the processed material to a special chemical form, but the major part of the charge will be based on the cost of reprocessing in an assumed plant whose operating cost (including interest and amortization on capital cost) is \$15,300 per day, and whose capacity is 1 ton of natural or slightly enriched uranium per day. It is possible that the capacity of the plant may be lower under certain conditions, for example, if the enrichment of the feed material is high enough to be subject to criticality limitations.

The announcement does not state a limit on definition of slightly enriched uranium, but since the intent is evidently to cover most large power reactors operating with uranium fuel one would guess that the limit would be not less than 3 percent enrichment. In addition to the charge for the actual processing time, an additional charge will be made to cover startup,

shutdown, and cleanup of the system between fuel batches. This charge will amount to not less than 3 days of operating time nor more than 8 days, and between these two limits will be equal to the processing time. A charge of 1 percent of the value of Government-owned source or special materials in the processed batch will be made to cover process losses.

It is inevitable that any guarantee of processing costs over a significant period of time will affect the over-all economics of nuclear power, and influence the direction of reactor development. It can be said that the establishment of a low processing charge will encourage the use of enriched fuels, whereas a high reprocessing cost would encourage a tendency toward the use of natural or very slightly enriched uranium.

As an example of the effect of reprocessing cost on fuel cost, we may consider a reactor using uranium of 2 percent enrichment and operating with a conversion ratio of 0.70. For a reasonably large reactor of this type which would require processing of enough fuel to keep the end losses of processing low, one might conclude that the guaranteed reprocessing cost would be something like \$20,000 per ton. The fraction of the fuel cost (mills/kwh) attributable to reprocessing will then depend upon the exposure of the fuel in the reactor, and will amount to 3.3 mills/kwh at 1,000 megawatt days per ton for a power plant of 25 percent efficiency. If a computation is made of the burnup of U^{235} in such a reactor and the net production of plutonium for various irradiation times, it is found that net cost of the fuel burned (cost of U^{235} burned minus credit for plutonium in reactor) does not vary greatly with the irradiation time. It decreases slowly from approximately 1.5 mills/kwh at zero to approximately 1 mill/kwh at 20,000 megawatt days per ton.

To determine the effective fuel burnup cost, one must add to this net cost the cost of reprocessing. At 10,000 MWD/ton, this will amount to 0.33 mills/kwh and at 20,000 MWD/ton, to 0.17 mills/kwh. If the cost of reprocessing a ton of fuel is independent of enrichment, then

the reprocessing cost in mills per kilowatt hour will also be independent of enrichment for a given degree of irradiation (MWD/ton) of the fuel. It may also be expected that the net cost of the fuel burned will not be a very sensitive function of fuel enrichment whenever the enrichment is more than about 1½ percent. Thus, for reactors which require considerable enrichment for criticality, it will probably be advantageous to load with enrichments high enough to assure that fission damage to the fuel, rather than loss of reactivity, is the limiting factor in fuel life.

Despite the relatively low reprocessing cost which is indicated in the Commission announcement, reactors using fuel enrichments in the neighborhood of 2 percent or more must have quite good neutron economy and high conversion ratio to approach a fuel burnup cost (including reprocessing) of 1 mill/kwh or less. In this connection, the proposal of W. B. Lewis for long burnup, natural uranium reactors is of interest (see Heavy Water Moderated Reactors, pp. 38-40).

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REACTOR PHYSICS

Lattice Theory

The three important reactor characteristics—criticality, conversion ratio, and reactivity lifetime—are mutually interrelated through the neutron physics of the reactor. Since long

term reactivity changes have a strong influence on nuclear power costs, it is of great importance that reactor theory be sufficiently accurate to give reasonable estimates of these reactivity changes, if not from fundamental cross section data, at least from measurements on critical and near critical assemblies, combined with observations of the reactivity changes which result from limited periods of operation. It is therefore a very practical requirement that reactor theory be capable not only of predicting criticality, but that it also give an accurate picture of the details of the neutron economy.

A redefinition of the fast fission factor ϵ has recently been proposed by B. I. Spinrad.¹ He points out that the probability of escape of fast fission neutrons from the fuel lump where they are formed may be substantially less than unity, and proposes that ϵ be defined not in the usual way (the number of neutrons making first collision at energies below the fission threshold of the fertile material per neutron born in thermal fission), but rather as the number of neutrons making first collision with moderator per neutron arising from thermal fission. The reference gives methods for computing the fast effect in U²³⁸ lattices under the latter definition. The result of this formulation is to predict a lower value of ϵ and a higher value for the fast fission contribution to the conversion ratio. It is stated in the reference that the result is to improve the correlation between reactivity measurements and conversion ratio measurements in D₂O-moderated lattices.

Spinrad's definition of ϵ has been used by E. Critoph² along with a revised method of computing resonance escape probability in a comparison of theory and experiment for D₂O-U lattices. The method of calculating resonance escape, which takes into account the variations in epi-resonance flux across a reactor cell, postulates that the U²³⁸ resonances—insofar as this flux distribution is concerned—can be represented by a single resonance at an effective energy E_R. It is necessary to assign a value for E_R, and this is done by empirically choosing a value (100 ev) which gives good agreement with the experimental Laplacians

measured on Zeep. When this method of calculating p is used in combination with Spinrad's calculation of ϵ , it is stated that the agreement between measured and calculated conversion ratios for NRX is greatly improved.

Resonance Integrals of Fertile Materials and Their Temperature Coefficients

Although enough critical and exponential experiments have been done to give some confidence in the semi-empirical methods of calculating resonance escape for uranium metal, the body of experimental data is meager for oxides and almost nonexistent for thorium and thorium oxide. Consequently, incremental information on resonance escape for the latter materials is of considerable importance. M. V. Davis³ has measured resonance integrals for U, UO₂, Th, and ThO₂, in cylinders of 1.73 cm radius by the reactivity coefficient method in the Hanford Test Pile. The measured values for p and for the effective resonance integral are as follows:

	p	$\int (\sigma_{\text{res}})_{\text{eff}} \frac{dE}{E} (\text{barns})$
U-----	0.872	10.87
UO ₂ -----	0.921	17.99
Th-----	0.918	11.10
ThO ₂ -----	0.926	13.77

The temperature coefficient of the resonance integral is important not only in determining the amount of reactivity which must be built into a power reactor for fuel temperature changes, but also in determining how much safety against runaway may be expected from the prompt temperature coefficient of the reactor. G. W. Rodeback⁴ has measured the temperature coefficients of the effective resonance integrals of U²³⁸ and Th by comparisons of their epicadmium neutron activations over a range of temperatures. The measurements were made on heated uranium and thorium cylinders of $\frac{1}{4}$ -inch diameter, near the reactor end of the SUPO thermal column. The surface-to-mass ratios of the cylinders were 0.117 cm² per gram for uranium and 0.189 cm² per gram for thorium. The measured values for the

temperature coefficient of the epicadmium activation were:

Uranium: $+0.8\% \pm 0.2\%$ per 100°C .
measured from 20° to 660°C .

Thorium: $+2.0\% \pm 0.3\%$ per 100°C .
measured from 20° to 775°C .

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HEAT TRANSFER: CALDER HALL TRANSVERSE FINS

Qualitative Discussion

In order to obtain reasonable heat fluxes in gas-cooled reactors, extended heat-transfer surfaces are generally used. Many configurations of extended surfaces are possible, as for example, longitudinal continuous fins, longitudinal strip fins arranged in line or staggered in successive fin rows, and pin or spine-type fins. In the Calder Hall reactor, the heat-transfer from the uranium slugs to the pressurized CO_2 gas-coolant is accomplished through the use of transverse (or circumferential) fins as illustrated in the sketch at the top of figure 3. The fin material is a magnesium alloy (Magnox) which has a high thermal conductivity value.

Of the many fin types and shapes possible, practically no work has been conducted on the

transverse fin (with axial flow) outside of the recent British effort. The reason for this general lack of interest has been the idea that the gas could not "find" its path into and out of the narrow fin spaces in sufficient quantity compared to the "bypass" flow to give good heat-transfer. In addition it would appear that the turbulence generated would result in large pressure drops. In view of the successful application of such fins in Calder Hall, it appears worthwhile to examine their applicability and relative performance for gas-cooled reactors in general.

The experimental work by the British on a large variety of fin types and shapes led them to the conclusion that transverse fins are superior to other types for the Calder Hall reactor design. Pictures of the gas-flow pattern in the fin spaces show that the flow conditions lead to the creation of two coupled vortices in the fin spaces. (See photograph in reference 1, page S17.) The transfer of momentum and mass (and hence, of heat) between the gas "pocket" in the fin spaces and the main flow stream evidently takes place through the mechanism of these eddying vortices.

The character of the eddy formation in the flow of a fluid over blunt bodies depends quite critically on the Reynolds number. For example, for flow over spheres and cylinders, the flow pattern behind the body can vary from a turbulent irregular wake to one characterized by a periodic formation and shedding of well-defined vortices depending on the Reynolds number for the flow. The same dependence on Reynolds number of the flow pattern over a given transverse fin configuration is to be expected. It would appear, then, that the relative merits of transverse finning in comparison with other fin types would depend on the mass-velocity requirements and size of the individual finned-tube unit required in the application.

To illustrate, in the Calder Hall reactor, the gas flows over the transverse fins at an average speed of about 40 feet/sec. at a gas pressure of about 115 psia. From many standpoints in power reactor design, it is desirable to increase

the mass velocity of the gas through the reactor by increasing in gas pressure (and hence, gas density) and gas velocity. The experimental information on the present Calder Hall transverse fin design indicates that the performance drops off quite rapidly with increase in Reynolds number beyond the present application. Hence, a re-evaluation of fin types would be necessary for higher mass flows.

Of equal importance is the consideration that if gas temperatures and heat fluxes are increased, fin heights must be decreased, even with materials of high thermal conductivity such as aluminum and magnesium. Further, slug diameters would be decreased to avoid excessive center temperatures, and grouping of the slugs might be required for reasons of reactor physics and mechanical design. Quite

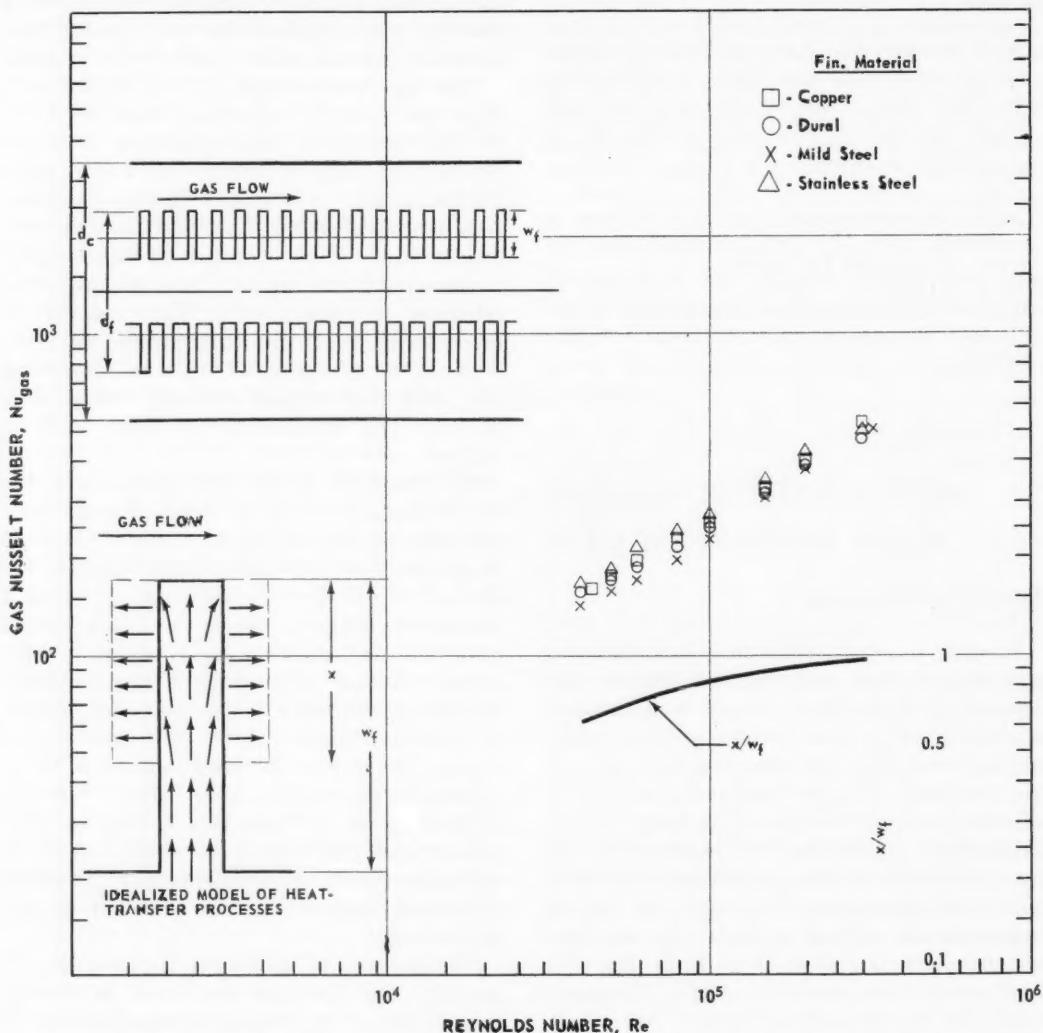


Figure 3. Gas Nusselt Number and fraction of fin length "scrubbed" by gases for different fin material. (Calculated from data given in reference 1.)

evidently such changes of configuration would affect the choice of fin arrangement.

In order to facilitate the use of the available data for evaluation of transverse fin performance under different conditions, an analysis has been made, in "Analysis" below, of the published experimental data. The analysis bears out the conclusion that the relative performance of transverse fins is quite sensitive to geometrical and flow conditions, and that for the particular conditions characteristic of the Calder Hall reactor they appear to be definitely superior to other types. The data do not cover the range of interest for higher performance reactors, but there are indications that the transverse arrangement will be less attractive for such reactors.

Analysis

The following section presents an attempt to obtain a more basic understanding of the heat-transfer processes involved in heat flow from circumferential fins. The heat-transfer and heat-transfer-to-friction ratio for a circumferential fin configuration are compared with those for smooth and roughened surfaces. In reference 1 (p. S18, fig. 6) heat-transfer experimental data are presented for a particular transverse-fin configuration. In plot (A), Stanton number is plotted versus Reynolds number for air using four different fin-materials—copper, dural, mild-steel, and stainless steel. Reference 2 (pp. 323 and 328-329) gives a list of symbols applicable to the data of reference 1 and to the discussion presented herein. It is important to note that the heat-transfer coefficient h is associated with the hypothetical perimeter of the outer fin tips (πd_f from figure 3). Also, the equivalent diameter d_e of the flow passage is $(d_o - d_f)$. The heat-flow path from the base of the fins to the gas is indicated in the sketch at the bottom of figure 3. It is obvious that fin-material thermal conductivity is important in the determination of the heat-transfer coefficient as defined. This dependence of the defined heat-transfer

coefficient on fin material is shown in reference 1 (p. S18, fig. 6). It is shown, however, that the data separation can be effectively eliminated by the use of $(Re)(\frac{k}{km})$ instead of Re (Reynolds Number) in the plot.

In the following section, results are presented of an analysis of the data in reference 1 made on the basis of the idealized model of the heat-transfer processes shown in the sketch at the bottom of figure 3. In this model, heat is conducted through a fraction of the fin width ($\frac{1-x}{w_f}$) transferring no heat to the gas coolant. For the remaining portion of the fin width (x/w_f), the heat is conducted through the fin and transferred, at a rate associated with a constant heat-transfer coefficient h_{gas} , from the fin surface to the gas. The fin-tips are treated as part of the fin surface in the analysis. Note that h_{gas} is the heat-transfer coefficient between the gas and that fraction of the fin surface from which heat is assumed to be transferred and so does not include the fin heat-conduction processes. This is in contrast to the defined heat-transfer coefficient h used by the British (reference 2, pp. 323 and 328-329), which is really an overall heat-transfer coefficient for the fin configuration.

For a given transverse fin geometry and flow Reynolds Number, the value of x/w_f and of the gas Nusselt Number $Nu_{gas} = h_{gas} d_e / k$ should be independent of fin material and type of gas coolant. This conclusion involves the very reasonable assumptions that (1) the effective eddy diffusivity of momentum for the flow process is a function only of the Reynolds Number, and (2) the relation between the effective eddy diffusivities of heat and momentum is independent of the gas.

Using the ideal model of the heat-transfer processes involved as shown in the sketch of figure 3, it is a simple matter to convert the overall measured heat-transfer coefficient to the gas coefficient, h_{gas} . For each Reynolds Number the value of x/w_f was determined which, for the different fin materials used in the tests, gives essentially the same value of h_{gas} . Figure

3 contains a curve of x/w_f versus Re obtained by this procedure. Also plotted in figure 3 are the values of Nu_{gas} for the different fin materials. The points for the different fin materials fall together, certainly within the accuracy of heat-transfer measurements; the small spread in points shows no trend with fin thermal conductivity value. Considering the range of thermal conductivity values covered (max.-to-min. value of 18.5:1), the fact that the above procedure brings the points for the four different fin-materials together shows that the ideal model assumed herein for the heat-transfer processes involved is a reasonably good one.

Figure 3 shows that Nu_{gas} and x/w_f both increase with Re . At $Re=4 \times 10^4$, $x/w_f=0.6$; that is, the effective length of the fins "scrubbed" by the gas in the fin space is 60 percent of the total fin width. At $Re=3 \times 10^5$, $x/w_f=0.94$, indicating that nearly the entire fin width is "scrubbed" by the gases. The Nu_{gas} points plotted show that Nu_{gas} increases with about the 0.56 power of the Reynolds Number. This value compares to the 0.8 power for smooth tubes and about the 0.9 power for roughened tubes. Since x/w_f is nearly 1 at $Re=3 \times 10^5$, no further gain in heat-transfer performance through this factor is possible with increase in Reynolds Number. Also, the increase in Nu_{gas} with Re is at a rate much lower than for smooth or roughened tube surfaces. Finally, fin effectiveness will decrease with an increase of heat-flux. Hence it is to be expected that the performance of the particular configuration should decrease quite rapidly with further increase in Re above 3×10^5 . This points up the need for redesign and reoptimization of fin geometry for increased mass-velocity requirements.

Figure 4 presents heat-transfer data obtained from reference 3 on a particular transverse fin configuration (with Magnox as the fin material) for 3 different channel diameters d_e . In figure 4, Nu (defined as hd_e/k) is plotted versus Re . In addition, lines are plotted giving the heat-transfer characteristics for flow over a smooth surface and over a roughened surface; these heat-transfer characteristics are obtained from

reference 4. Again, it must be pointed out that different types of surfaces cannot be evaluated relative to one another on the basis of heat-transfer and friction characteristics without a specific application in mind. However, it is convenient here, for comparison purposes, to consider the smooth and the roughened surfaces to be tubes of diameter equal to the fin-tip diameter d_f over which the gas flows in the annular space between the tube and the outer can. With this geometry, the heat-fluxes for given mass flow conditions are related to Nu differently for the different types of heat-transfer surface. Figure 4 shows that at $Re=10^5$, the heat flux obtained for the transverse fin configuration with $d_e=4$ inches is 2.2 times larger than with the roughened tube and 3.5 times larger than with the smooth tube. The drop in performance of the transverse fin-geometry with increased mass flow is clearly indicated in the figure; extrapolation indicates that at a $Re=7 \times 10^5$, the roughened tube (with no fins) gives the same heat-transfer as the transverse fin design.

Figure 5 compares the heat-transfer-to-friction ratio for the same configurations treated in figure 4. In figure 5, $(St \times Pr^{0.6} / \frac{f}{2})$ is plotted versus Re where Pr is the Prandtl Number of the gas, St is the Stanton Number, and f is the flow friction factor. For the smooth tube, this ratio is equal to unity. For the transverse-finned tube with a 4-inch diameter channel, the heat-transfer-to-friction ratio is 0.86 at a Reynolds Number of 10^5 . This means that slightly higher pumping powers per unit of heat transferred would be required for the transverse finned tube than the smooth tube. However, it is remarkable that the heat-transfer-to-friction ratio approaches so close to the smooth tube value inasmuch as most attempts to increase heat-transfer coefficient by turbulence promotion or mixing cause the ratio to drop quite drastically. In the case of the British transverse fins 3.5 times the heat-transfer rate for a smooth tube is obtained at a very small increase in pumping power. Note that for the roughened tube, which gave only about 1.6

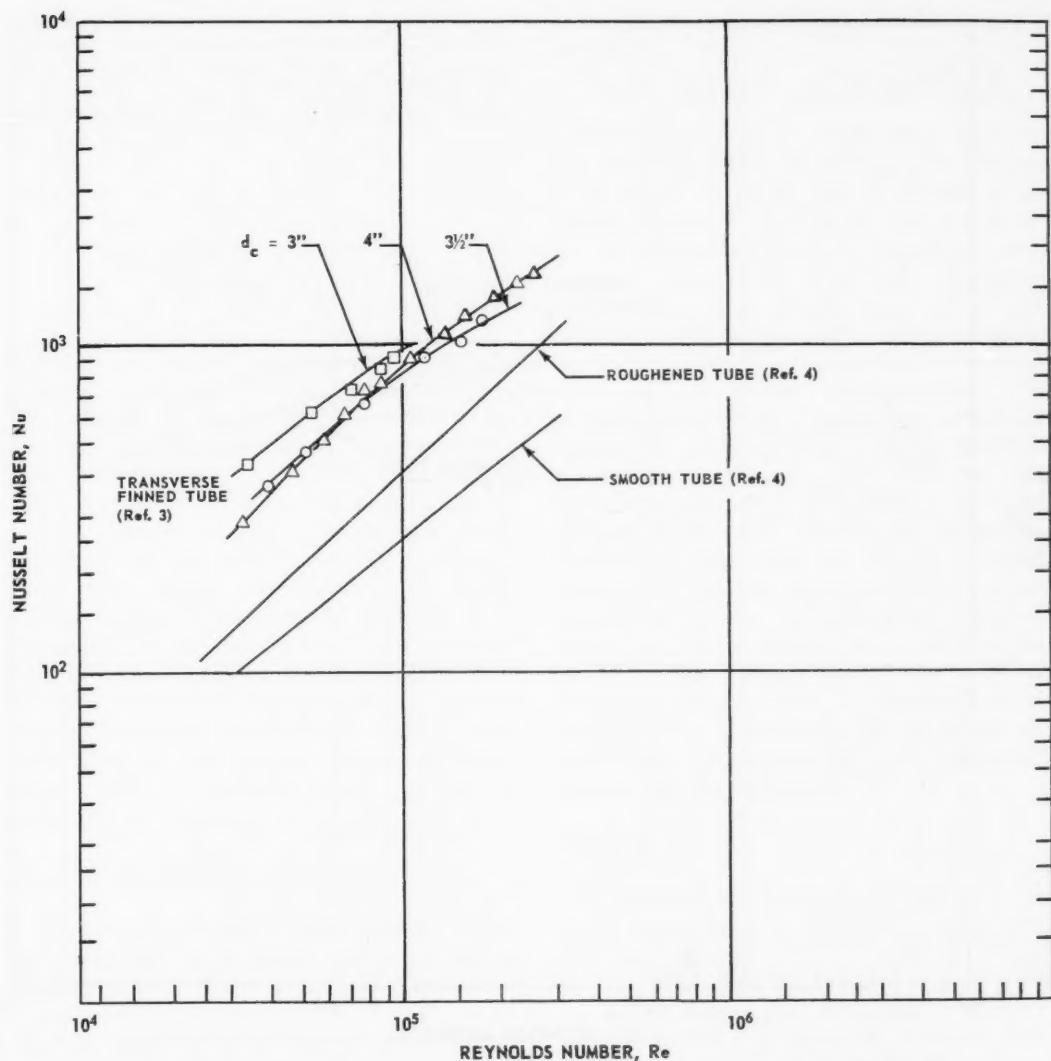


Figure 4. Comparison of heat-transfer characteristics of transverse-finned tubes with smooth and roughened tubes.

times the heat transfer for a smooth tube, the heat-transfer-to-friction ratio dropped to 0.54 for $Re=10^5$. Figures 4 and 5 point out the large advantages of the British transverse fin design for Reynolds Numbers in the range of 10^5 .

Again, attempts to increase mass flow (and hence Reynolds Number) into the range of

Reynolds Numbers about 10^6 apparently result in a less favorable ratio of heat-transfer to friction. At the 2.5×10^5 Reynolds Number value, which is the upper limit of the data, the ratio has dropped to 0.73 and the curve trend indicates faster dropoff beyond 2.5×10^5 .

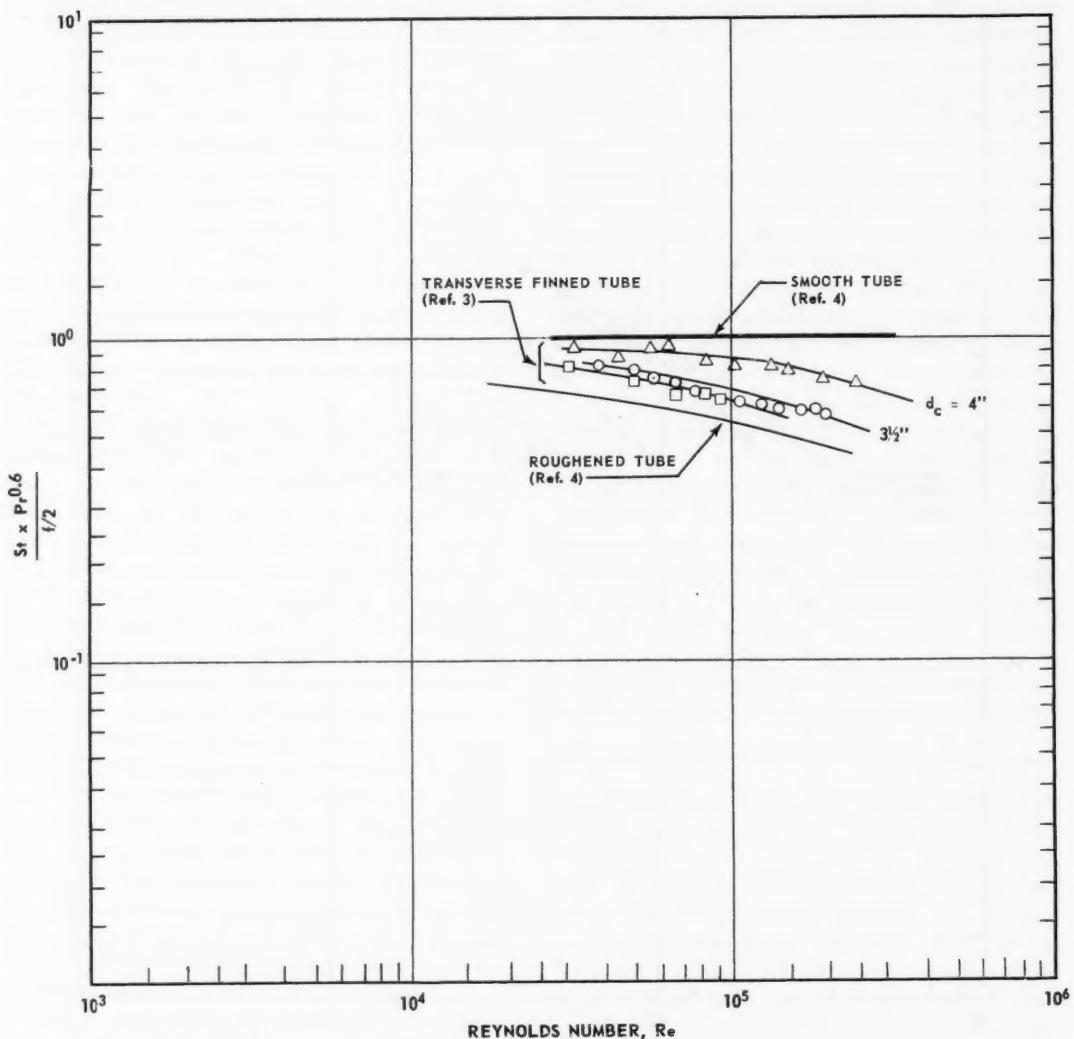


Figure 5. Comparison of ratio of heat-transfer to friction for transverse-finned, smooth, and roughened tubes.

In view of the excellent performance of the British transverse fins in the Calder-Hall range and of the interest in power reactor designs to increase mass velocity through the reactor and possibly to go to smaller finned-tube designs, further experimental effort on transverse fin performance beyond that presently available from the British work will be worthwhile.

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REACTOR KINETICS AND CONTROL

Questions related to reactor stability are presently of interest in connection with the fast breeder reactor and the boiling water reactor. It is notable that an experimentally-discovered instability in the EBR-1 reactor has occasioned considerable concern as to the safety of such reactors, whereas boiling reactors have been built and are being built with the knowledge that regions of instability do exist. Insofar as this difference in the implications of instability for the two reactor types is a real one, it indicates that there are other factors involved in addition to instability. The following discussion attempts to define these differences in the process of summarizing recent work on stability which appears in references 1, 2, and 3.

Power Coefficients of Reactivity

Whenever it is possible for a change in the operating power of a reactor to cause a change in the reactivity, the reactor is said to have a power coefficient of reactivity. It can also be said that under these circumstances the reactor is part of a feedback loop. If an increase in power causes an increase of reactivity, the power coefficient of reactivity is said to be positive, or alternatively the loop is said to have positive feedback; a decrease in reactivity as a result of a power increase is characterized conversely by a negative coefficient or a negative feedback. The power coefficient of reactivity is conveniently expressed as the change in reactivity produced by a given change in power divided by the magnitude of the power change,

expressed as a fraction of normal operating power, P_0 :

$$\text{Power coefficient of reactivity } \frac{\Delta k}{k} / \frac{\Delta P}{P_0}$$

Ordinarily the power coefficient of reactivity is caused by power-induced changes in temperature, density, or position of one or more of the reactor materials. It may be inherent in the reactor proper or may also involve changes in coolant temperature which are propagated around the entire coolant loop. It is quite possible, and in fact probable, that in a complex reactor or reactor system a number of coefficients may be present simultaneously. The characteristics of these coefficients which are important in determining stability are their magnitudes and the speed with which they come into operation following a change in reactor power. These characteristics can be specified conveniently by imagining that the reactor power is oscillated sinusoidally at constant amplitude over the entire range of frequencies which may be of interest, and specifying the amplitude and phase lag of the resultant reactivity change over the spectrum of frequencies. Such a specification of amplitude and phase shift is often referred to as the transfer function for the coefficient.

Stability

Unless some kind of reactivity feedback exists in the reactor system, either because of a power coefficient of reactivity or because an artificial control system has been applied to the reactor, the reactor system does not constitute a closed servo loop, and it is impossible to speak of the stability or instability of the system. It can be said, however, that in such a case the reactor is in a state of unstable equilibrium. That is to say, the power will be constant with time only if the reactor is exactly critical—a situation which cannot be attained in practice. Practical examples of such reactors are reactors which are operated at such low power that temperature effects are very small. The power of such a reactor will always in time drift farther and farther from its initial value unless

an operator or an automatic control system is provided to make occasional small adjustment of reactivity. If the reactor does have some kind of power coefficient of reactivity the system becomes a closed loop which may be stable or unstable. If a single coefficient exists, and if it is negative, the reactor will tend to be self-regulating, although the regulation may be poor if the coefficient is small. The system may be stable or unstable depending upon the magnitude of the coefficient and its phase lag.

In general an increase in the magnitude or an increase in the phase lag of the coefficient increases the tendency toward instability. It is impossible to specify in a qualitative way the limits of stability. Little and Schultz in reference 1 have summarized quantitatively the conditions for stability for a particular reactor model. In the case of the stable reactor with negative feedback the self-regulating property may or may not be sufficiently effective to simplify the job of reactor control; but it does not usually increase the difficulty of the job. If the reactor with its negative feedback is unstable, the power, if not otherwise controlled, will oscillate with ever increasing amplitude and with a period which is determined primarily by the time constant of the negative coefficient. In most cases such an instability will increase the difficulty of reactor control. It will not necessarily make the reactor uncontrollable if the natural frequency of oscillation is low; the degree to which it adds to the reactor hazard can vary from almost zero to a large factor, depending on the degree of instability and the availability of other forces to limit the amplitude of oscillation.

A reactor which has only a positive power coefficient of reactivity is always unstable. It has been customary to speak of such a reactor as autocatalytic. The "runaway" of such a reactor is not oscillatory, but consists of a power increase of ever increasing rate. It is quite feasible to control a reactor having a positive coefficient of moderate size. The presence of the positive coefficient, however, may increase quite seriously the consequence of a reactor "runaway", if one does occur in spite of the

operation of the control system. The seriousness of this effect will certainly vary with the individual reactor design; but other things being equal, it may be expected to be particularly serious for the fast reactor, in which a small change of reactivity in the vicinity of prompt criticality can produce a very large change in the reactor period.

Behavior of EBR-1

The case of EBR-I is complicated by the experimental evidence of both a positive power coefficient of reactivity and an instability of the oscillatory type. Briefly, the experimental evidence was as follows:

1. The positive coefficient was observed by stopping the coolant flow to the reactor and adjusting the control rods at low power to give a positive exponential period of about one minute. The power was allowed to increase, with the control rods stationary, through values in the normal operating range. As the power became high enough to give appreciable temperature increases, the period became progressively shorter, indicating that reactivity was increasing. It is to be noted that this positive coefficient was observed quantitatively only in this case of no coolant flow, which is not a normal operating condition. There was some evidence of the positive coefficient after sudden changes in coolant flow with otherwise normal operation, but no quantitative measurements could be made.

2. Oscillations of reactor power were observed in a few instances when the reactor was operated with reduced coolant flow. When the reactivity was oscillated sinusoidally by the motion of a control rod, power resonances were observed at certain characteristic frequencies. These frequencies were in the range .03 to 0.1 cycles/sec. and depended on rate of coolant flow, being lower for lower coolant flows. These experiments have been summarized by Brittan in reference 4.

From the beginning, certain aspects of the EBR-1 behavior have been puzzling. It has not been possible to demonstrate the existence of a phenomenon which will explain the positive power coefficient. The possibility that it is due to Doppler broadening of U^{235} fission resonances has been considered, but the experimental and theoretical evidence is that this phenomenon could not account for a coefficient of the observed magnitude. It has been postulated that the coefficient is due to bowing of the fuel rods because of the net temperature gradient which exists across the diameter of any given fuel rod. Theoretical analysis indicates that an effect of the observed magnitude could conceivably result, but there is no direct evidence that the bowing actually exists.

The second puzzling feature of EBR-1 behavior is the relatively low value of the resonant frequency. The fact that this reactor or any other reactor should show resonances is not particularly surprising, but it has been impossible up to this time to find a source of negative power coefficient with a time constant of the same order of magnitude as the period of resonance. The fact that oscillations of coolant temperature cannot feed back around the sodium loop (since there is a sodium reservoir of large volume in the circulating system) confines the possibilities to the internals of the reactor itself. It is postulated that thermal expansions of parts of the reactor structure lead to a negative power coefficient of long time constant, but no specific mechanism has been demonstrated.

A number of theoretical studies of the reactor oscillation have been made. The most comprehensive and most elegant of these has been made by Bethe.³ This study considers not only the possible phenomena in EBR-1, but also the question of how instabilities can be predicted in other reactors. The conclusions are quoted below.

"1. Any reactor which has a negative delayed power coefficient is apt to have a resonance, i. e., at some power P_1 , free oscillations with frequency ω_0 may take place.

"2. In the presence of a delayed negative coefficient, the resonance will *always* occur if the prompt power coefficient is positive, but it may also occur if it is negative. A sufficiently large negative prompt power coefficient, however, will suppress the resonance.

"3. When the reactor power is increased beyond the 'resonance' power P_1 , the reactor will begin to show spontaneous oscillations of power whose amplitude will grow exponentially with time. It will then be intrinsically unstable.

"4. To find out whether a resonance exists, either oscillator tests or *accurate* observations of the transients after a sudden change of control setting (transient tests), are satisfactory.

"5. Details of the resonance, particularly its sharpness, may be observed by either method. Such experiments also give information about the damping of oscillations below the critical power P_1 , and the rate of their exponential increase above P_1 .

"6. Complete and accurate information on the power coefficient as a function of frequency can be obtained only from the oscillator method, not from the transient method. Such information is useful for devising a theoretical model of the power coefficient for the given reactor, and thus to assess likely causes for its undesirable features, if any, with a view to improving the design of the reactor.

"7. Either type of test may be carried out safely.

"8. Either type of test permits the prediction of a resonance from the result of tests at quite low power (typical figures are 1/4 to 1/2 of the critical power P_1).

"9. A positive 'prompt' coefficient is only of minor importance for the stability of the reactor in normal operation, and of minor influence on the resonance. However, the positive prompt coefficient is very undesirable in case of an accident started by other causes, unless the total reactivity increase which can occur due to this mechanism is limited to a small amount.

"10. The prompt power coefficient can be deduced only rather inaccurately from observations on the resonance. It can best be measured by oscillator experiments at low coolant flow. Experiments of the large excursion type (as carried out on EBR-1) are unsafe, not very revealing, and unnecessary.*

"11. EBR-1 has a positive prompt and a negative delayed coefficient, in agreement with the analysis of Kinchin.

"12. The 'slow' component of the power coefficient is presumably related to the transfer of heat to the reactor structure. In EBR-1, most of the time involved in this transfer is consumed in convection of heat rather than conduction.

"13. The time scale of the 'fast' component of the power coefficient is determined by the heat capacity of the reactor core and has nothing to do with the neutron generation time.

"14. In line with 13, neither the resonance nor the prompt positive temperature coefficient of EBR-1 is a direct consequence of this reactor working with fast neutrons."

By analyzing the shape of the reactor resonance curve, Bethe finds that the combination of a positive power coefficient of 0.8×10^{-3} with a time constant of approximately 1 second, and a negative coefficient of -3.8×10^{-3} with a time delay of 10 seconds, would account for this shape. These would correspond to temperature coefficients of 0.6×10^{-5} per degree C. and -3.0×10^{-5} per degree C., respectively. Thus, by mathematical analysis, a reasonably consistent description of the reactor dynamics can be given in terms of reactivity coefficients. The explanation of these coefficients—especially the negative one—in terms of the physical structure of the reactor remains to be accomplished. It is understood that experiments for this purpose will be made with a new EBR-1 core by Argonne National Laboratory.

It might be remarked in passing that if positive and negative coefficients of more nearly

equal magnitude are postulated, the magnitude of the time delay necessary to give the observed resonance frequency can be considerably reduced. It is probable, however, that this combination does not give the observed shape of the resonance.

With respect to the problem of investigating the dynamic behavior of an existing reactor, Bethe's paper demonstrates effectively that oscillation experiments are the most powerful method of attack, and that they will suffice for testing the stability of a reactor and for determining the existence of positive power coefficients of reactivity. It should be emphasized, however, that there is a long step between the determination of the transfer function of a reactivity coefficient and the explanation of the coefficient in terms of the physical behavior of the reactor parts, a step which must be taken before the designer can proceed with confidence to a new reactor design. For the accomplishment of this step—which was the objective of the EBR-1 tests—other types of experiment may be quite important, particularly if they include observations on other quantities, such as local temperatures, in addition to the reactor power variation. Further, it should be recognized that certain limitations of the oscillation technique exist which are not immediately apparent. The most important of these perhaps is the circumstance that no method exists for determining the reactivity worth of a control rod at operating power in a reactor which has unknown power coefficients of reactivity. Thus, any information which is derived from the absolute magnitude of the reactor "gain" is subject to an inherent uncertainty.

The Boiling Reactor

The possibility of instability in the boiling reactor has always been recognized, and in fact it was the anticipation of such instability that delayed serious work on boiling reactors until a relatively recent date. The initial Borax tests, whose purpose was to investigate the stability characteristics, showed that a given boiling reactor can be operated stably up to

*In deference to the experimentalist's point of view, it might be said that when the location of the reactor is suitable, they are quick, direct, and convincing.—Editor.

some power level characteristic of the reactor, and that at higher power levels operation becomes unstable. It seems reasonable that for a group of boiling reactors having the same general characteristics, the criterion for stable operation will be the amount of reactivity compensated by steam. The experiments which have been made to date tend to confirm roughly this hypothesis: evidences of instability have begun to appear when the amount of reactivity compensated by steam is something like 1.5 percent k_{eff} and instability usually occurs by the time this reactivity is as great as 2.5 percent. As a corollary, it may be said that, other things being equal, boiling reactors which have a small power coefficient of reactivity will operate stably at higher power densities than will reactors with lower power coefficients of reactivity.

The natural circulation boiling reactor has the following characteristics which influence its dynamic behavior:

1. The power coefficient of reactivity has only a negative component. This situation is provided for in the design of the reactor, and although it is true that at least one reactor has been built for which the formation of steam near the center of the core produced a positive reactivity effect, the effect was a minor one, and one which could not give a large total reactivity increase.

2. The negative power coefficient is relatively large and has a relatively short time constant. In fact, the boiling reactor typically has quite effective self-regulation of its power. Typical power coefficients of reactivity range from -0.01 to -0.02 . The coefficient is determined by the neutron-physics characteristics of the reactor, and is susceptible to adjustment by adjustment of the physics design; but it has proved difficult to keep the coefficient as low as 0.01 in a practical design.

3. There is a relatively large "noise" component in the reactivity feedback because of fluctuations in the steam content of the reactor. It is possible that this "noise" contributes to the eventual instability.

4. Although there is quite strong mutual coupling between average power and the average value of reactivity (the averages being taken over a period of a few seconds), there also exists the possibility of reactivity fluctuations which result from the hydrodynamics of the system, and which are coupled only very loosely to reactor power. There is evidence that such fluctuations are the major contributors to the eventual instability of the reactor.

In the boiling reactor, the time constant associated with the negative power coefficient of reactivity appears to be the thermal time constant of the fuel plate. The time constant was of the order 0.1 second for the Borax reactors and is of the order 0.5 second for EBWR. With such a rapid response, the boiling reactor would be expected to be stable even for very large values of the power coefficient of reactivity, provided the analysis based on small reactivity fluctuations is applicable. (This is the theory which has been applied to the fast reactor, and which is generally applicable, provided all of the forces which may perturb the reactivity are relatively tightly coupled to the reactor power.) The boiling reactor eventually reaches a point of instability because the reactivity fluctuations fail to satisfy this requirement. When the operating power of the boiling reactor is so high that the steam content of the core corresponds to a reactivity of two or three times the delayed neutron fraction β , the noise fluctuations and the loosely coupled hydrodynamic oscillations begin to have the capability of changing the reactivity by an amount comparable to β . Under such excitation, the time response of reactor power is no longer that of a reactor with the normal complement of delayed neutrons, but begins to approach that of a reactor without delayed neutrons. For this case, the time constant of the fuel plate is no longer short compared to the time response of reactor power, and the possibility of instability exists.

The experimentally observed behavior in the Borax reactors was that when the reactor power became sufficiently high, an oscillation of

reactivity, of approximately sinusoidal shape, of frequency roughly one cycle per second, and of varying amplitude, became apparent. It seems reasonable that this oscillation is caused by a gravity oscillation of the body of water in the reactor, but no proof of this hypothesis has been obtained. The amplitude of the oscillation grows and decays for no apparent reason, although the average amplitude is higher for higher power levels. If the power level is high enough, the amplitude may at some time become so large that the condition for stability is exceeded. The amplitude of the oscillation then increases fairly rapidly with each oscillation until a new regime of operation occurs which is characterized by large steady oscillations of reactivity of roughly constant amplitude. The power fluctuations associated with this reactivity fluctuation are generally large—the ratio of maximum/minimum power may be of the order 100 to 1,000. This type of operation has been called *chugging*. The "noise" amplitude and the tendency toward oscillation were observed to decrease as the operating pressure was increased.

In the Borax reactors, the power instability was not a hazard because even though the peak powers reached during the chugging operation were quite high they were not sufficiently high to damage the reactor. Indeed, it had been shown for those reactors that the effectively instantaneous addition of 2.1 percent excess reactivity could be tolerated without damage to the reactors. It was therefore quite evident that no instability could be damaging to the reactor provided the total reactivity represented by the steam content was not more than 2.1 percent. In actual fact, it was found that the total available reactivity was never observed to be effective in any individual chug; that is to say, there was no evidence that during the chugging operation the reactor core was ever completely empty of steam. This characteristic of the reactor, which determines how violent the chugging may become if instability ever should occur, is a question quite separate from the question of stability, although it will depend fundamentally on the same

general characteristics of neutron lifetime, fuel plate time constant, hydrodynamics, and operating pressure.

The recently reported SPERT instability² appears to be essentially the same phenomenon as the chugging observed in Borax. The circumstance that the instability followed a ramp addition of reactivity is probably more or less incidental. It should be mentioned in this connection that in the case of the Borax reactors the mere operation of the reactor under conditions which had previously led to instability did not always guarantee chugging, but rather that there were certain operating conditions under which instability might eventually become apparent if operation were continued for a sufficiently long time. Under these conditions, rapid changes of operating conditions, particularly externally-induced changes of reactivity, could trigger the instability. The SPERT instability did occur at a somewhat lower value of reactivity (1.5 percent minimum) than was the case in Borax, but not so much lower as to be particularly surprising, especially in view of the shorter neutron lifetime in the SPERT reactor. It appears also that in the SPERT case a larger fraction of the total available excess reactivity appeared during the individual chugs.

Significance of Instabilities

It may be said in summary that the possibility of instability is not confined to certain reactor types, but neither is an unstable characteristic to be expected frequently in the design of new reactors. When the possibility of instability does exist, it is something to concern the reactor designer, but its significance for reactor safety can be defined only in relation to other dynamic characteristics of the reactor. In the specific case of the fast reactor, it is important to distinguish between the significance of an instability, which simply increases the difficulty of maintaining control of the reactor by the operator or by an automatic control system, and the significance of a positive component of the power coefficient which,

if large enough, could increase importantly the severity of any hypothetical reactor runaway.

Evaluation of the significance of these characteristics is colored also by the circumstance that it has been more difficult to set limits on the possible energy release in the case of a fast reactor runaway than in the case of accidents to some other reactor types. It is possible that some information on this subject may be obtained from detailed examination of the damaged EBR-I core.⁵ It should be emphasized that the stability or instability of a given reactor is determined by its specific design and not by its general type. Furthermore, as Bethe has pointed out, it is quite feasible to determine by safe experiments whether a reactor, after it is built, has regions of instability, and whether it has positive power coefficients of reactivity.

In the case of the boiling reactor, instability is due to quite different causes, and in fact, the reactor is stable so long as the power level is kept below that at which large uncoupled reactivity fluctuations can occur. Confidence in the safety of this reactor is also based to a considerable extent on the experimental observations that instability, when it does occur, is preceded by characteristic signs which give ample warning, and by the fact that the boiling reactor has a high degree of tolerance for power excursions. It must be remembered, however, that this tolerance for excursions is very much a function of the specific reactor design and cannot be assumed to be the same for reactors with low conductivity fuel plates as for the early Borax reactors.

In this connection, the Borax-IV experiments which are being conducted by Argonne National Laboratory with fuel elements of ThO_2-UO_2 are of great interest. It is understood also that the feasibility of predicting instability in the boiling reactor by the oscillator technique is being investigated on this reactor. It may be that such predictions cannot be very precise because of the very nature of the boiling reactor, but if they prove to be practical at all, they will be of considerable value, since it is not clear at the present time how one determines

the safe power limit of operation for a boiling reactor, other than by approaching relatively close to that limit.

Finally, in connection with the boiling reactor, it should be pointed out that the stability limits observed to date appear to be characteristic of the natural circulation case, and that there is no experimental evidence as to what the limitations will be for forced circulation. It would be surprising if the forced circulation boiling reactor did not have somewhat broader regions of stability than the natural circulation reactor.

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SAFEGUARD AND CONTAINMENT

Out-of-Pile Tests on Metal-Water Reactions

In an interim summary report,¹ Aerojet General Corporation reported further results of tests to determine the extent and violence of chemical reactions between fuel materials and water.

This series of tests dealt with uranium, uranium-molybdenum, uranium-zirconium-niobium, zircaloy-B, and aluminum. In some tests a 1-inch diameter stream of molten metal was simply poured into water. In other tests

a small blasting cap was used to disperse the molten metal stream in the water.

No explosive reactions occurred when there was no dispersion of the molten metal stream even though the percentage of metal reacting was high in some cases (10–30 percent for uranium at 1540° C.). When the metals were dispersed with a blasting cap, the percentage of reaction was large. The rates of pressure rise and the magnitude of the maximum pressures were considerably less than those obtained from high explosives. However, shock waves of magnitudes which would represent severe containment problems for reactors were observed. Percentages of reaction observed for dispersed molten metals were as follows:

Metal or alloy	Estimated percentages of reaction (average)
Uranium	45
U-Mo	40
U-Zr-Nb	70
Zircaloy-B	75
Aluminum	75

The violence and completeness of reaction were quite sensitive to droplet size. For example, for zircaloy-B the percentage of reaction was 5 percent or less for droplets of 2,500 microns diameter as compared with 75 percent reaction for particles 100 microns in diameter. Observed variations in reaction with particle size for zircaloy-B could be explained by an assumption that oxidation consistently progressed to a depth of 25 microns.

Metal temperature appears to be second in importance only to droplet size as a factor in explosive reactions. For all metals tested except aluminum, violent reactions on dispersed metal occurred at temperatures only slightly above the melting point. In a series of tests with dispersed molten aluminum at 930° C. the percentage of reaction varied from 12 to 50 percent, with no explosion observed except for the 50 percent reaction case in which a slight explosion was noted. In one test with aluminum dispersed at 1565° C., a heavy explosion involving

75 percent reaction was observed. It should be noted that the melting point of aluminum is low (660° C.) compared with the other metals tested and it is not surprising that comparable explosions were obtained when the aluminum temperature was raised to near the melting points of the other metals.

Addition of small percentages (0–2 percent) of beryllium to zircaloy-B in an effort to reduce the severity of explosions gave erratic results. The authors state that beryllium shows some promise as a reaction retardant.

In-Pile Tests on Metal-Water Reactions

Tests were conducted in the MTR to determine the reactive properties of fuel elements containing uranium in aluminum, Nichrome V, and zirconium in a water environment.² Small test samples containing sufficient U²³⁵ to cause melting were cut out of fuel plates and placed in ceramic cups in autoclaves. Irradiation was carried out in the high flux region ($2-4 \times 10^{14}$) of the MTR. The autoclave pressure and sample temperature were recorded during each run. Gas analyses were made after each test.

A summary of the in-pile test results is shown in table V-1.

A reaction between metal and water occurred in all tests as indicated by the formation of hydrogen gas. No pressure transients were observed except in those cases where the maximum temperature of the samples exceeded the melting points. For aluminum samples, maximum pressures of 8,000 and 23,000 psi were recorded in two runs in which the maximum sample temperature was about 1,400° F. At temperatures in excess of 2,200° F. no pressure transients were observed even though the percentage of reaction went as high as 70 percent. This is not easily explained and is in contradiction with other (out-of-pile) tests¹ in which the violence of reaction increased with temperature.

In general, samples of uranium-zirconium clad with zirconium gave higher percentages of reaction than the aluminum elements, although a pressure transient (10,000 psi maximum) was observed in only one case. A series of moderate

TABLE V-1.—IN PILE METAL-WATER REACTION TESTS (COMPILED FROM TABLES IN IDO-16257)

Run number	Approximate sample weight		H ₂ gas found mol%	Maximum recorded temperature °F.*	Maximum rate of temperature rise °F./sec	Time to maximum recorded temperature sec	Time to first pressure transient sec	Time to maximum pressure transient sec	Maximum pressure transient psi	Percentages of complete reaction	Remarks	
	Total gms	U ²³⁵ gms										
<i>Specimens—(U-Al) Al Clad</i>												
(U-Al) Al Clad	1	0.066	0.015	2.8	—	—	—	—	—	0.3	Data may be doubtful on gas, since several weeks passed before the gas analysis was made.	
	3	.066	.015	1.9	—	—	—	3.75	4.0	0.3		
	4	.066	.015	7.0	—	—	—	—	—	0.9		
	5	.066	.015	1.4	1,400	580	8.15	6.35	6.35	23,000		
	6	.066	.015	4.5	—	—	—	—	—	0.2		
	7	.066	.015	7.0	—	—	—	—	—	0.4		
	23	.066	.018	11	2,600	—	0.80	—	—	—	5.1	
(U-Ni-Cr) Nichrome V Clad	24	.064	.018	66	2,200	5,100	0.00	—	—	—	52.8	(UO ₂ -Ni-Cr) Nichrome V Clad
	25	.073	.027	69	2,300	5,100	0.48	—	—	—	70.2	
	27	.052	.012	6	3,000	2,100	0.51	—	—	—	2.4	
	33	.062	.017	11	2,700	2,800	0.91	—	—	—	7.5	
	9	.144	.021	—	—	—	—	—	—	—	—	
	10	.098	.184	—	1,800	700	3.6	—	—	—	—	
	11	.339	.151	29	2,200	700	7.3	—	—	—	3.4	
(U-Zr) Zr Clad	12	.45	.196	64	2,100	900	2.0	—	—	—	15.8	(U-Zr) Zr Clad
	13	.47	.101	69	2,800	7,000	2.59	10.8	10.91	18,000	14.3	
	14	.43	.092	65	2,200	2,400	3.28	—	—	—	12.6	
	15	.40	.085	87	2,500	2,500	1.87	—	—	—	10.9	
	29	.09	.015	7	—	—	—	.4	.4	1,600	6.3	
	30	.42	.089	75	3,800	2,700	1.86	—	—	—	33.4	
	28	.14	.021	12	2,800	1,700	7.42	—	—	—	4.5	
(U-Zr) Zr Clad	32	.13	.028	3	3,800	2,400	1.82	1.0	2.0	3,000	2.9	(U-Zr) Zr Clad
	16	.20	.083	80	2,800	4,500	0.73	—	—	—	82.2	
	17	.20	.079	75	4,000	7,000	0.75	—	—	—	86.9	
(U-Zr) Zr Clad	18	.20	.079	67	4,000	3,400	4.15	—	—	—	80.8	(U-Zr) Zr Clad
	19	.40	.161	67	5,700	26,000	0.63	17.3	18.22	10,000	17.4	
	20	.20	.061	51	3,300	6,000	0.65	—	—	—	16.4	
	26	.21	.025	—	2,500	1,600	0.76	—	—	—	—	
(U-Zr) Zr Clad	31	.20	.036	79	3,900	5,700	0.55	—	—	—	63.9	

* Since in most cases the thermocouple used (Iron-Constantan and Chromel-Alumel) melted or may have alloyed with the fuel, the tem-

peratures indicated are generally qualitative rather than quantitative.

reaction bursts, rather than a single large one, seemed to be the characteristic behavior, perhaps indicating that a protective oxide film formed intermittently, which subsequently ruptured to bring fresh metal into contact with water again. Analysis of reaction products indicated that both the zirconium and uranium constituents of the fuel samples reacted with water.

A discrepancy is noted between references 2 and 3 regarding the zirconium element reaction tests. In reference 2 the pressure surge of 10,000 psi is connected with test number 18, which had a maximum rate of metal

temperature rise of 3,400° F./sec. (maximum recorded temperature 4,000° F.). In reference 3 the 10,000 psi pressure was associated with run number 19, which had a maximum rate of temperature increase of 26,000° F./sec. (maximum recorded temperature 5,700° F.).

All Nichrome V samples also reacted with water to form hydrogen. Pressure transients were observed in 3 of the 11 tests. The maximum pressure of 18,000 psi occurred in a test whose reaction was 14 percent complete. From thermodynamic considerations no explosive reaction should occur between nickel and water. The explosive reaction presumably was between

chromium and water and/or conversion of UO_2 to U_3O_8 .

Practically all of the materials which are commonly considered for use in fuel elements for thermal reactors have, upon occasion, reacted violently with water.² Results thus far are somewhat inconsistent and in some cases apparently contradictory. The problem of obtaining practically useful answers to the safety questions raised by the possibility of explosive metal-water reactions is a formidable one. The two important questions are: (1) Is it ever possible to guarantee that such reactions will not occur in the event of melting of metal fuel elements or fuel jackets in a water-cooled reactor? (2) If not, what are the conditions of energy release and pressure which must be handled in the containment design? It would appear that much remains to be done before these questions can be answered.

Theoretical Study of Metal-Water Reactions

A fairly comprehensive treatment of possible mechanisms influencing the chemical reaction of metals with water is given in references 4 and 5.

Pressure and Temperature Transients in Reactor Containment Building

Experiments are described in the reference 6 which indicate that the maximum pressure which will result in the containment building from rupture of a water moderated reactor pressure vessel will be less by a factor of about two than values calculated on the assumption that no heat is absorbed by shell walls and surfaces of internal equipment.

A tank having a volume of 792 feet³, 470 feet² of surface area, and capable of withstanding an internal pressure of 45 psig was used for the tests. Various amounts of water (up to 292 pounds) were heated in an external pressure tube and injected into a stream distribution pipe within the tank through a quick opening valve. Water temperatures corresponding to saturation pressures up to 1,000 psig in the boiler tube were used. A motion picture camera

recorded synchronous readings on pressure and temperature gauges during the transient tests.

Significant observations and conclusions made by the authors include the following:

1. The maximum containment shell pressure usually occurred within 7 seconds from the instant of opening the valve.
2. Temperature readings on the external surface of the steel shell increased rapidly during the first 8–10 seconds, after which the rise was very slow. In a typical case the heat absorbed in the shell during the first 10 seconds was 51,000 Btu as compared with only 6000 Btu during the next 10 seconds. The heat transfer from the vapor-air mixture to the steel shell apparently starts at a very high rate (600–1,100 Btu/hr, ft² ° F.) and decreases rapidly. In the 10–30 sec. interval the heat transfer rate is between 50 and 100 Btu/hr. ft² ° F.
3. A concrete liner 1½ inches thick inside the steel shell reduced the heat absorption in the walls about 40 percent compared to the bare shell. In two tests the amounts of heat absorbed by the bare shell and concrete lined shell during the first 10 seconds were 110 Btu/ft² and 66 Btu/ft², respectively.
4. In one experiment which was analyzed in considerable detail the maximum pressure observed was 24 psig. The calculated pressure for the hypothetical case of no heat absorption in anything other than air was 50 psig.
5. When the observed results are applied to the EBWR reactor and containment shell, it is concluded that the maximum pressure which would result from rupture of the pressure vessel containing 20 tons of water at 488° F. (The EBWR actually contains only 10 tons. This amount was arbitrarily doubled to take into account additional energy from a metal-water chemical reaction.) would be only 5–7 psig as compared with the design value of 15 psig.
6. In one test, a tray containing 590 pounds of water was placed under the steam distribution tube within the shell. The tray was arranged so that the steam discharged would

create a water spray effect. No pressure rise at all was observed during this test.

The results cited in this paper show that previous pessimistic assumptions of no heat absorption in structures following rupture of a reactor pressure vessel within a containment shell lead to expensive overdesign of the containment building. The authors suggest further studies of this type on a larger shell, and believe that by proper arrangement of cold-water containers around the reactor, it may be possible to practically eliminate all pressure rise in the shell.

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FUEL ELEMENTS

General Discussion of Fuel Element Characteristics

Most of the published work on fuel elements relates to the manufacture of elements or to the testing of elements for the purpose of predicting their lifetime in a reactor. Recent work of this kind is summarized later in this section. It is useful to consider first, however, the relationship of various fuel element characteristics to reactor performance, and the significance of element lifetime. This discussion applies to

fuel elements for large central-station power reactors.

1. Relation of Fuel Element to Fuel Cost—No generalized quantitative treatment of this subject can be given because specific fuel cost (mills/kw-hr) is affected strongly by other characteristics of the reactor, as well as by the fuel element. A qualitative discussion is, however, useful.

It is useful to break the specific fuel cost down into three components:

a. Use charge

This is a capital charge against the value of the fuel in the inventory necessary for operation of the reactor. It is applied against the fuel in the reactor, against the fuel in the process of fabrication, and, when reprocessing is used, against fuel being reprocessed and awaiting reprocessing. If government financing of the fuel inventory is used, the use charge is currently 4 percent of the value of the fuel. Quite evidently the factors favoring a low use charge are: inexpensive fuel (e. g. a U^{235} - U^{238} mixture of low enrichment—see figure 1); high specific power; low inventory in fabrication and reprocessing; and high net thermal efficiency.

b. Fuel burnup charge

This is an operating expense equal to the net value of fuel destroyed per kilowatt hour of electricity produced. The net value of fuel destroyed is computed as the value of the fuel fed into the reactor minus the value of the fuel recovered by reprocessing after reprocessing charges have been applied against it. The factors favoring low burnup charge are: use of cheap fuel; high conversion ratio (to give a large amount of salable product and long reactivity lifetime); long fuel lifetime (to reduce the effect of reprocessing cost); and high net thermal efficiency.

c. Fuel fabrication cost

The cost of fuel fabrication results in a capital charge against the fuel inventory and

an operating charge for replacement of fuel elements discharged from the reactor. Quite obviously, a low unit cost of fabrication favors a low figure for both the capital and operating charges. Long fuel lifetime reduces the operating fabrication charge in inverse proportion, and reduces the capital charge to some extent because it reduces the ratio of total inventory to inventory in the reactor. Both charges are, of course, inversely proportional to the net thermal efficiency. It is not entirely obvious what the unit should be to which the fabrication cost and fuel lifetime are applied. However, the specific charge per kilowatt hour is unaffected by the choice of unit provided consistent units are used for fabrication cost and fuel lifetime. It is convenient to express fabrication cost in dollars per pound of fuel (fertile plus fissionable isotopes) and lifetime in megawatt days per ton of fuel.

It is clear that to minimize fuel cost for any given type of fuel element one must strive for long fuel lifetime (both with respect to fission damage, corrosion, etc., and with respect to reactivity), low fabrication cost, and maximum neutron economy. It is not so obvious, however, how one can make comparative ratings of different fuel element types with respect to these characteristics. In most reactor designs which have been proposed, all three components of the fuel charge listed above make a significant contribution to the total, and it can be said that the fuel element characteristics which tend to minimize these components are always important ones.

2. Fuel Element Lifetime—As pointed out above, long fuel lifetime is important in reducing both fabrication and reprocessing costs. If the reprocessing cost is a fixed price per ton of fuel, unaffected by enrichment in the range of interest, then the significant unit for fuel lifetime is megawatt days per ton of fuel. If the reprocessing cost is a fixed price per gram of fissionable isotope in the fuel, then the fuel lifetime expressed as fissions per initial atom of fissionable isotope is the significant one affect-

ing the specific reprocessing charge. Frequently, the fuel lifetime, or the duration of an irradiation test, is expressed in percentage burnup of the fuel atoms. For a mixture of U²³⁵ and U²³⁸ metal the relation among various units of lifetime is:

$$1\% \text{ burnup of fuel} \approx 0.84\% \text{ of fuel atoms fissioned}$$

$$1\% \text{ burnup of fuel} \approx \frac{.01}{E} \text{ atoms "burned" per initial fissionable atom}$$

$$1\% \text{ burnup of fuel} \approx \frac{.0084}{E} \text{ atoms fissioned per initial fissionable atom}$$

$$1\% \text{ burnup of fuel} \approx 6,950 \text{ MWD/T}$$

where

$$E = \text{initial enrichment} =$$

$$\frac{\text{atoms of fissionable isotope}}{\text{total fuel atoms}}$$

and the useful energy release per fission has been taken as 190 Mev. The relations for UO₂ and for ThO₂-UO₂ mixtures are very nearly the same if 1 percent burnup of fuel is interpreted to mean the burnup of 1 percent of the heavy atoms. If it is interpreted to mean 1 percent of all the atoms present (including oxygen), then the numbers on the right side of the above set of equations must all be divided by 3.

3. Absorption by Additives—From the standpoint of reactor physics the ideal fuel element is one which contains only "fuel"—i. e. fissionable and fertile material. Practically, the addition of other materials is often necessary for improvement of corrosion performance, irradiation lifetime, or other characteristics. The neutron absorption by such additives must be kept low if good neutron economy is to be maintained. In particular, if the fuel elements are to be used in natural uranium reactors, the absorption by additives must be very low. If enriched fuel is used, the parasitic absorption by additives can be compensated, at a price, by additional enrichment.

The effect of parasitic absorption on neutron economy can be visualized by considering the expression for the maximum possible internal conversion ratio (R_c) of a reactor fueled with U^{235} and U^{238} .

$$R_c = \frac{\eta_{25}\epsilon}{1+\tau B^2} - R_{fp} - R_r - R_p - \left(1 + \frac{0.00407}{E}\right) L^2 B^2 - 1$$

where η_{25} = number of neutrons produced per absorption in U^{235}

ϵ = fast fission factor

τ = slowing-down area

L^2 = diffusion area

B^2 = geometric buckling

E = enrichment = atoms U^{235} /atoms
($U^{235} + U^{238}$)

and R_{fp} is the ratio of absorption by fission products to absorption by U^{235} , R_r is the corresponding ratio for absorption by any parasitic control elements which are in the reactor core during operation, and R_p is the corresponding ratio for all other parasitic absorbers in the reactor. The average room-temperature macroscopic absorption cross section of U^{235} in natural uranium is approximately 0.20 cm^{-1} . Thus, the addition of a parasitic absorber to the fuel element, in an amount sufficient to have an average macroscopic cross section of 0.01 cm^{-1} , would result in a decrease of 0.05 in the conversion ratio of a natural uranium reactor. The addition of the same density of the absorber to a fuel element having twice the natural enrichment ($E=0.0143$) would decrease the conversion ratio by only 0.025.

4. Fuel Density in the Element—The density of fuel in the element (i. e. number of grams of fissionable plus fertile material per cm^3 of fuel element) affects neutron-physics performance in several ways, and in general it can be said that the optimum physics design of the reactor will change if large changes are made in fuel density (e. g. if oxide elements are used instead of metal elements). The effects are most pronounced in the natural uranium reactor. The most important effect is the increase in surface/mass ratio for a given fuel-element

size. This increase causes a higher relative parasitic absorption in fuel cladding, process tubes, etc. It also requires, for a given resonance escape probability, that a lower mass ratio of fuel to moderator be used, and thus results in higher relative absorption by moderator. Finally, it results in a lower fast-fission effect for a fuel element of given size and shape. These effects, plus some increase in neutron leakage which results from the lower average scattering cross section in the reactor, can reduce considerably the maximum value of the material buckling which can be attained with natural uranium.

When enriched fuel is used, the above effects on neutron economy are less important, since they can usually be compensated by increase of enrichment coupled with rearrangement of the fuel lattice to give the appropriate resonance capture. Nevertheless, the effects cannot be discounted. Usually the effect of fuel density on the performance of a given reactor type can be evaluated only by detailed calculation supplemented by some design work.

In reactor types, such as the boiling water type, which place restrictions, other than those arising from considerations of neutron economy, on neutron leakage and resonance escape, the effect of fuel density may be especially pronounced, and may have important effects on the reactor design.

Uranium Metal and Alloys of High Uranium Content

Fuel elements of these types are applicable to large power reactors using U^{235} as the initial fissionable isotope and U^{238} as the fertile isotope. They have the advantages of high density and reasonably high thermal conductivity. Their nuclear performance is excellent provided the absorption cross section of alloying materials is kept small. Important physical constants of the pure uranium metal are as follows:

Grams of fertile plus fissionable material per cm^3 —18.5–19.0 g/cc wrought, pressed and sintered

Thermal conductivity—0.074 cal/cm² sec (°C/cm) at 600° F.

Melting point—1,133° C.

Temperature of α — β phase transition 663° C.

Temperature of β — γ phase transition 764° C.

Recent development and testing of elements of this kind is summarized below. In cases involving additives, an average macroscopic parasitic absorption cross section (Σ_p) is given for the additives. This is computed on the assumption that the alloy has a density of 18.5 g/cc rather than its true density. This cross section can be compared directly with the macroscopic cross section of uranium to give an estimate of the effect on neutron economy, as illustrated in the section "3. Absorption by Additives," p. 22.

1. Natural Uranium—Two specimens of unalloyed uranium (1.3 inches diameter x 4 inches long), containing a 0.375 axial hole and clad in 0.050 inch thick Al and insulated from the clad by 0.004 inch of aluminum oxide, were irradiated under the following conditions:

Burnup—0.091–0.108% of total atoms

Surface temperature—480° C. to maximum 1,000° C.

Operating power—16.6–17.2 watts/gm of U

Examination after irradiation showed no warping or external evidence of dimensional instability. However, the axial hole in each specimen had closed uniformly over the length of the element, the average density of the metal changing from 18.9 to 17.7 g/cm³.^{1e}

2. Uranium-Zirconium [Parasitic absorption cross section: $\Sigma_p = (0.01949 \text{ cm}^{-1})(\text{w/o Zr})$]—Argonne National Laboratory reports the optimum alloy with respect to radiation damage as U-1.6 w/o Zr. Here the growth rate is 8–10 percent per 1 percent burnup of total atoms at low irradiations with an average growth of 5 percent at 1 percent burnup.

Test specimens of U-2 w/o Zr have shown good dimensional stability at high burnups at temperatures less than 600° C.^{1g}

3. Uranium-Zirconium-Niobium [Parasitic absorption cross section: $\Sigma_p = (0.01949 \text{ cm}^{-1})(\text{w/o Zr}) + (0.1169 \text{ cm}^{-1}) (\text{w/o Nb})$]—U-5 w/o Zr—1.5 w/o Nb, the reference fuel for the first loading in EBWR, is suitable for exposures to moderate burnups at moderate temperatures. However, the excellent aqueous corrosion characteristics of this alloy is almost completely lost by the heat treatment to obtain radiation stability.^{1g}

4. Uranium-Niobium [Parasitic absorption cross section: $\Sigma_p = (0.1169 \text{ cm}^{-1}) (\text{w/o Nb})$]—Specimens of U-3 w/o Nb were tested and found to have poor radiation resistance.^{1g}

U-10 w/o Nb specimens operated up to 800° C. with 0.69 a/o burnup showed density decreases as large as 25 percent. However, a U-10 w/o Nb fuel assembly (10 plates) defected with a 0.040 inch hole exposed to 0.1 a/o burnup showed very little corrosion in a water loop with the temperature at the defect of 700° F.^{1b}

5. Uranium-Molybdenum [Parasitic absorption cross section: $\Sigma_p = (0.2573 \text{ cm}^{-1}) (\text{w/o Mo})$]—U-5, 7, 10 w/o Mo alloys have been checked and U-10 w/o Mo alloy was found to have the best radiation stability. The specimen of U-10 w/o Mo was tested at a specimen temperature of 240° C. and a burnup of 0.12 total a/o.¹¹

6. Uranium-Silicon [Parasitic absorption cross section: $\Sigma_p = (0.0475 \text{ cm}^{-1}) (\text{w/o Si})$]—The best U-Si alloy tested was the U-3.8 w/o Si or U₃Si homogenized delta phase alloy which was tested at burnups of 0.71 percent of uranium atoms and central temperatures of 280–860° C. The specimens were found to have excellent dimensional stability and preliminary indications show good aqueous corrosion resistance.^{1g}

Uranium-Plutonium Alloys

The U-Pu alloys below have compositions of interest for fast breeder reactors.

U-20 w/o Pu extruded specimens irradiated to 0.7 total a/o burnup showed good dimensional stability.^{1g}

Uranium-Plutonium-Fission alloy (U-20 w/o Pu—5 w/o Fs) shows good radiation stability at 1 total a/o burnup. Specimens containing U-20 w/o Pu—3 w/o Fs showed signs of dimensional instability.^{1g}

Fission is a mixture of materials having metallurgical properties similar to those expected for the fission products in recycled fast-breeder fuel. It has the following composition (Fs—4.6 w/o Zr, 25.9 w/o Mo, 39.8 w/o Ru, 6.5 w/o Rh, 23.2 w/o Pd).

Metallic Thorium and Thorium-Uranium

Recent experimental results on these materials are the following:

Long term tests of Th and a series of Th-U specimens containing 0.1, 1.0, and 5 w/o U-235 irradiated to burnups of 3.5 percent of total atoms at "moderate" temperatures showed dimensional changes of less than 2 percent with no changes in the surface.^{1g}

Four aluminum clad thorium slugs (6.5 inches long x 1.4 inches diameter) were irradiated to exposures of 2×10^{20} — 6×10^{20} nvt thermal. Dimensional changes were of the order of —0.010 inch on the diameter.^{1a}

Several plates having a 0.020 inch thick x 0.75 inch wide core of 98.7 w/o Th-1.2 w/o U-235 were irradiated in MTR to an integrated flux of 2.6×10^{21} neutrons/cm². The plates used 52-S Al clad and a 1.3 percent Si-Al bond. The specimens were dimensionally stable, core hardness approximately doubled, and production of U-233 was as much as theory would indicate.^{1a}

Non-metallic Elements of High Fuel Content

1. UO_2 —Some of the significant properties of UO_2 are:^{2, 3, 4}

Theoretical density—10.97 g/cm³

Density of fissionable plus fertile material (theoretical)—9.7 g/cm³

Melting temperature— $2,760 \pm 30^\circ \text{C}$.

Lattice constant— $5.472 \pm 0.005^\circ \text{A}$

Coefficient of thermal conductivity k for a body having a density 10.02 g/cc:

$100^\circ \text{ C}—0.018 \text{ cal/cm}^2 \text{ sec } (\text{°C}/\text{cm})$

$600^\circ \text{ C}—0.008 \text{ cal/cm}^2 \text{ sec } (\text{°C}/\text{cm})$

$1,000^\circ \text{ C}—0.006 \text{ cal/cm}^2 \text{ sec } (\text{°C}/\text{cm})$

Linear coefficient of thermal expansion—

$11.2 \times 10^{-6}/^\circ \text{C}$.

Specific Heat—0.056 cal/g/° C. at 17° C .

Commercially available MCW-Urania compacted and sintered ceramic elements have bulk densities of 10.44 g/cm³ (approximately 95 percent of theoretical density).

UO_2 is relatively inert chemically, can operate at high temperature, and is believed to be relatively insensitive to fission damage. Against these advantages, it has the disadvantages of brittleness, relatively low fuel density, and low thermal conductivity. The latter disadvantage is in some measure compensated by the high permissible temperature; but oxide fuel elements, for a given heat flux, will probably be thinner than metal elements. This characteristic, combined with the low density, usually causes a significant loss in neutron economy relative to that which can be attained with metal elements.

The question of the degree to which fission products will escape from oxide elements is an important one. Preliminary measurements of fission gas release from UO_2 gave the following results:

In porous (65 percent dense packed) UO_2 packed in a S. S. can, approximately 40–50 percent of the fission gases are released. Some solid fission products are also released. In 150 specimens tested no radiation induced failures have occurred.^{1e}

In compacted and sintered ceramics of 70–95 percent of theoretical densities, the gas released during irradiation of 5–10 mgm amounts of UO_2 for 25 MWD was less than 1 percent. In 3–4 hours at 800° C . after irradiation approximately 1–2 percent was released. When oxygen was present as a trace impurity in the helium sweep gas, then approximately 30 percent was released.⁵

Irradiation studies on various samples of UO_2 indicate that the fission gas release is a strong function of the fractional porosity, the fission gas release observed in 8.0 percent

porosity elements being an order of magnitude greater than a specimen tested having a 3.5 percent porosity.

Analysis of gas release on a specimen of MCW 91 percent of theoretical density irradiated to 3,000 MWD/T with an estimated central temperature of 1,600° C. showed a 3.1 percent gas release on puncture of the clad at room temperature with very little released (less than 0.2 percent) at temperature up to 1,000° C.^{1b}

Fission gas release studies⁶ on undefected hot water loop specimens of UO₂ ceramics canned in Zircaloy-2 indicate the following:

UO₂ ceramics of 95 percent of theoretical density and irradiated to as high as 5,000 MWD/T with maximum central temperatures <1,500° C. when punctured at room temperatures released less than 1 percent of the fission gases. The same ceramics operated at maximum central temperatures of greater than 1,500° C. released a substantial (> 20 percent) of the fission gases.

UO₂ ceramics of less than 95 percent of theoretical density showed an increase in gas released with decrease in density with 10 percent of one test gas Kr⁸⁵ being released from a specimen having 80 percent of theoretical density after 1.65 MWD/T burnup.

Recent results of irradiation tests on UO₂ fuel elements are the following:

Scattered physical properties measurement on UO₂ specimens clad in Zircaloy-2 and

irradiated to 5,600 MWD/T show no appreciable engineering defects. In pile cycling tests to burnups of 2,000 MWD/ton showed no dimensional changes or ratchetting except for a slight increase in length.⁵

Specimens of 93 percent of theoretical density showed no dimensional changes with irradiations to 0.9 a/o burnup and a maximum center temperature of 1,570° C.

Other specimens including a full length PWR rod (10 $\frac{1}{4}$ inches x 0.413) of EUR-UO₂ (93 percent of theoretical density) were defected with a 0.005 inch hole and irradiated at central fuel temperatures in excess of 3,000° C. The Zircaloy jacket on these specimens ruptured and examination of the specimen showed that melting had occurred over the entire length of the specimen and 3 concentric temperature zones were distinguishable.^{1b}

Although Zircaloy-2 clad UO₂ has been used successfully, tests have shown that zirconium at temperatures greater than 1,100° F. will reduce the UO₂ and become brittle.

The corrosion resistance of UO₂ bodies in 650° F. water and 650°-750° F. steam is excellent. Tests indicate that UO₂ is inert in water containing any measurable excess of hydrogen.⁵

The following table indicates somewhat the effect of high burnups on Zircaloy-2 clad UO₂ ceramics:⁶

Experiment number	Number of specimen	Enrichment, percent U ₂₃₅ in total uranium	Maximum thermal flux (nvt)	Approximate integrated flux (nvt)	Maximum burnup (MWD/T)	Maximum changes		
						Length (inches)	Diameter (inches)	Density (percent)
14-14 ⁴	4	0	3. 6x10 ¹⁴	85x10 ²⁰	37, 500	0	+ 0. 010	Not measurable.
14-18*	4	6. 24	1. 0x10 ¹⁴	18x10 ²⁰	25, 000	+ 0. 014	+ 0. 007	+ 2.8.
14-19**	4	0	4. 4x10 ¹⁴	23x10 ²⁰	16, 300	Not measurable.	+ 0. 015	Not measurable.

*Specimens were prepared to PWR specifications.

**Specimens were the same as 14-14, except the diametral clearance of

the pellets was increased to 0.009 inch from the 0.003 inch of the PWR reference clearance.

The Zircaloy-2 clad in these specimens operated at a temperature of 650° C. and could be easily deformed.

The UO₂ was probably melted in the high exposure rods of 14-14 and 14-18, but sectioning has not been completed. One specimen of the 14-14 group had a clad failure caused either by internal gas pressure or volumetric increase of the UO₂ on melting. There was essentially no gas release from the 14-14 specimens when punctured, but a small crack was found in the cladding of one of the rods indicating that it had become brittle and cracked under irradiation. The clad for the 14-14 specimens was manufactured in 1955 and is not representative of the quality now being used.

Examination of the 14-19 rods revealed that the UO₂ had melted over approximately one-half of the radius. The cladding of these rods was examined and no hydride formation was found, indicating the samples had not picked up hydrogen during irradiation and that relocation due to thermal diffusion of the hydrogen present in the clad prior to irradiation had not occurred.⁶

EXPERIMENTS ON ADDITIVES TO UO₂⁷

Improvements in corrosion and thermal-fracture resistance of a UO₂ ceramic were sought through the use of additions of beryllia, beryllium, ceria, silica, silicon, silicon nitride, or zirconia. Nuclear considerations dictated that the modified bodies contain at least 80 volume per cent of UO₂ after sintering.

Sound specimens meeting this UO₂-content requirement were made with the beryllia, ceria, silica, and zirconia additions. These ceramics, with the exception of UO₂-silica, had as good resistance to 650° F. water and 750° F. steam as the unadulterated UO₂ ceramic.

Adding refractory-grade beryllia to the UO₂ powder compacts improved the thermal-fracture resistance of sintered specimens by a factor of about 1.7. Readily sinterable beryllia powder did not give this benefit. Variations in sintering treatment appreciably affected the thermal-fracture characteristics of the UO₂-BeO specimens.

Minor additions of titania or ceria to UO₂-beryllia powder compacts resulted in increased densities at lower sintering temperatures, but thermal-fracture resistance was not improved through their use.

Dense, strong specimens were made using ceria as the additive to UO₂. Ceria was the most effective density promoter during sintering of the major additives studied. UO₂-ceria specimens, however, were only slightly better in thermal-fracture resistance than those of UO₂ alone.

Zirconia and UO₂ combinations had adequate density and strength, but the zirconia had a deleterious effect on the thermal-fracture resistance of UO₂.

UO₂-silicon specimens had low density and strength, presumably because of the formation of a reaction product. The reaction occurred on sintering in hydrogen, argon, or in a vacuum, and, to a lesser degree, on hot pressing. The reaction product was isolated, but not identified. It was unstable when resintered by itself and with UO₂.

Satisfactorily dense and sound sintered compacts were not produced with either beryllium or silicon nitride as the additive.

2. Thoria and Thoria-Urania Mixtures—

The following are the general properties of ThO₂:^{4,8}

Density—10 g/cc

Melting temperature—3,300° C. ±100° C.

Lattice constant—5.595° A

Coefficient of thermal conductivity k for specimen density=9.6 g/cc:

100° C—0.020 cal/cm² sec (°C./cm)

600° C—0.008 "

1,200° C—0.0076 "

Linear coefficient of thermal expansion—
 $9 \times 10^{-6}/^{\circ}\text{C.}$ (500°—1,000°C.)

Specific heat—0.058 cal/g/°C. at 50° C.

Since thorium is not fissionable, a fuel element containing ThO₂ will always contain also a fissionable isotope in some form. Hence, the properties of thoria-urania mixtures are of interest.

Like urania, thoria has a high melting point and relatively low thermal conductivity, is relatively inert chemically, and poses comparable problems of fission product containment. Its density is relatively low (although not much lower than that of metallic thorium), but its high absorption cross section nullifies many of the nuclear disadvantages of low density.

Nuclearly, the most important characteristic of the thoria fuel element is that it must be provided with a fissionable isotope in highly enriched form. Thus the reactor using such fuel elements must always be fed with expensive fuel (the net fuel cost may, however, be low if conversion ratio is high and reprocessing is cheap). Once this disadvantage is accepted, however, the reactor designer has more freedom in selecting the ratio of fissionable-to-fertile isotope than in the case where U^{238} is the fertile isotope, since the cost of the fissionable isotope (per gram) is independent of this ratio.

Other nuclear disadvantages are the low fast-fission effect in thorium and the long half-life of Pa^{233} , which causes a significant reduction of conversion ratio in reactors of high thermal-neutron flux. A nuclear advantage is the high η of the U^{233} produced by conversion.

Recent experimental results on thoria elements follow:^{8, 9, 12}

As a fuel material, U_3O_8 has been added to ThO_2 in mixtures containing up to 30 w/o U_3O_8 . The U_3O_8 was then converted to UO_2 . The mixture of UO_2 and ThO_2 form a continuous series of solid solutions with the solidus-liquidus extending from 2,875° C. for UO_2 to above 3,200° C. for ThO_2 .

Specimens of 97½ w/o ThO_2 -2½ w/o UO_2 apparent density 9.27 g/cc (8.13 percent porosity), have been irradiated bare, but packed in steel wool to prevent mechanical damage. Exposures of 0.95 total metal a/o burnup were obtained and specimens were free of visible damage, but contained microscopic cracks.

Thermal cracking or fragmenting was noted in specimens of 90 w/o ThO_2 -10 w/o UO_2 (9.2 g/cc density, 5.4 percent porosity) canned in Al-1 w/o Ni, bonded with lead and

exposed in water at 500° F. and 680-810 psi to 0.29 total metal a/o burnup.

Other specimens of 90 w/o ThO_2 -10 w/o UO_2 (10.06 g/cc density-1.3 percent porosity) lead or welding gas bonded to cans of Al-1 w/o Ni were taken to central metal temperature of 3,900° C. at burnups of 1.25 of metal a/o. In the lead bonded case, failure appeared to be due to gas pressure with subsequent hoop tensile stress fracture.

Specimens with central temperatures of less than 2,000° C. with burnup as high as 0.64 a/o metal burnup were intact, with some fragmentation and very slight dimensional changes. The specific power of the above elements was 240 cal/sec/cc and the heat flux=38.1 cal/sq cm sec.

3. Urania-Silicon-Silicon Carbide—These fuel elements have quite low fuel density. They are visualized, however, as usable without cladding. If this hope materializes, good neutron economy may be attained without high fuel density. Fission-product containment is, of course, the important question involved in such an application.⁴

Recent experimental results are as follows:^{1a}

Fuel plates have been prepared by pressing UO_2 and SiC and then dipping in molten Si to form a 0.040 inch thick body of SiC in a silicon matrix at the surface of the element. The plates (½ inch x ½ inch x 1½ inches) were exposed in helium at an average surface temperature of 1,400-1,750° F. and a power generation of 120 watts/sq in, to an exposure of 3×10^{19} nvt. The results indicate excellent dimensional stability.

Other ceramics such as Cr-Al₂O₃-UO₂, UC, and UC₂ are also under study, but no significant information has been reported.

Elements of Low Fuel Content

Most of the fuel elements discussed below are for purposes other than large power reactors. At the present stage of development of fuel reprocessing, it is not evident that fuel elements of low fuel content can lead to economic central-station power.

Recent experimental results follow:

1. Uranium-Aluminum—Alloys have been manufactured with a maximum of 23 w/o U-235 in Al and indications are that as high as 30 w/o U-235 in Al can be satisfactorily made. The behavior of these elements under irradiation has been reported extensively in the literature.¹⁰

2. Uranium-Zirconium—U-25 w/o Zr to 90 w/o Zr have been studied to 0.64 a/o burnup. The 40, 50, 60 w/o Zr showed maximum dimensional changes of 1-1.2 percent. The 75, 90 w/o Zr were distorted and roughened by burnups of 0.1 a/o.¹¹

U-8 to 40 w/o Zr alloys were stable below 600° C.¹²

3. Uranium-Magnesium Dispersion—Magnesium dispersions containing 66 w/o U and 78 w/o U clad in S. S. cans were irradiated to burnups of 10 percent total uranium, at a fuel temperature of 300-400° C. with a heat flux from the element of 147-226 watts/sq. cm. Post-irradiation examination showed no changes in length or diameter, no escape of fission gases into the coolant, and no growth of fuel into a void space left in one end of each element.¹³

Specimens of 50 percent by volume U in a Magnesium-0.4 w/o Si matrix were canned in Zircaloy cans and irradiated in MTR at a flux of approximately 2×10^{14} nv. The core temperature was about 390° C. and the power generation about 1,570 watts/cc of uranium metal or 780 watts/cc of fuel. Examination of samples irradiation to 0.1, 0.5, 1.0, and 2.0 percent total U-atoms burnup indicate this material is dimensionally stable, resistant to radiation damage, and has satisfactory fission product retention properties. However, it has poor corrosion resistance in high temperature water, but preliminary tests indicate excellent corrosion resistance in boiling (255° C.) diphenyl. The thermal conductivity of a 64 w/o U-Mg specimen was approximately 2.5 times that of uranium metal.¹⁴

Three samples of powder metallurgy U-Mg elements in Al cans have been irradiated to 1,000

MWD/T and 3,000 MWD/T. Microporosity appears to have developed and the compact cannot be removed from the capsule, possibly indicating some dimensional instability or the interdiffusion of Al and U and/or Mg to form a bond.¹⁵

4. Urania-Aluminum Dispersion—Al clad U-54 w/o UO₂ (20 percent enriched) elements of the type used for the Geneva Reactor have been irradiated to 23 percent U-235 atoms burnup. Visual examination showed the elements to be in good shape.¹⁶

5. Urania-Stainless Steel Dispersion—Specimens of S. S. clad-UO₂-73.84 w/o 304 S. S., 0.35 B₄C fuel plates were irradiated to approximately 30 percent U-235 burnup at a temperature of less than 100° C.¹⁷

Other specimens were irradiated to maximum core temperatures of 850° C. for 1,000-hour irradiation times.

In general the following results were noted:

a. Core hardness varied inversely with UO₂ particle size as a result of irradiation.

b. Elements containing UO₂ particle sizes less than 31-44 microns showed evidence of core cracking when subject to a $\frac{3}{4}$ -inch radius bend.

6. Urania-Dispersions in Other Metals—Specimens of UO₂ and such matrix materials as Cr, Ni, and Fe were tested in helium-pressurized containers to various burnups (11-41 percent uranium atoms) at various temperatures (490° F. to 1,080° F.). In general the results were the same as with the stainless steel dispersion element. However, the electrolytic iron matrix elements showed a greater tendency for core cracking and core-clad separation and increased hardness compared to the S. S. elements.¹⁸

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OTHER REACTOR MATERIALS

Control Materials

1. Boron—Continued effort is being expended on boron alloys and compounds which have shown a satisfactory radiation stability with high burnout.

a. Boron alloys of SS, Zr, Zircaloy-2, and titanium have been irradiated to 40–60 percent burnup of the B¹⁰ atoms. All alloys tend to become brittle and lose their ductility with less than 1 percent burnup. Analysis of dimensional changes indicates a correlation which holds for these alloys.^{1e}

$$\frac{\%L}{\%A_b} = K \quad \%A_b = \% \text{ of total atoms burned (poison and matrix)}$$

K = a constant $\approx 0.8-0.9$ for these alloys

b. Stainless steels containing $\frac{1}{2}$ and 1 percent B¹⁰ were irradiated to approximately 10^{20} nvt. Dimensional checks indicate a growth in these specimens of approximately 1–2 percent.^{1a, p. 29}

c. A boron compound, zirconium diboride, was irradiated to 80 percent burnup of B¹⁰ atoms without fragmentation and with only 0.02 percent of the theoretical He content released after heating the specimen for one week at 700° F.^{1f}

d. Boron carbide hot pressed and tapped powder type were irradiated to burnup of 36 percent of B¹⁰ atoms. At 4 percent B¹⁰ burnup the hot pressed boron carbide releases 2.5 percent of the He, while tapped powder boron carbide releases 6.5 percent of the He. Increasing the burnup of the hot pressed boron carbide to

36 percent of B¹⁰ atoms increases the He release to 5 percent.

As the burnup increases, the physical damage increases, with complete granulation of the hot pressed disks occurring at 36 percent burnup of B¹⁰ atoms.^{1e, p. 30}

2. Cadmium—A liquid control rod¹ is proposed for power reactors using a liquid cadmium manometer having a helium reservoir in thermal connection with the reactor coolant.

3. Rare Earth Dispersions—Gadolinium Oxide (21.2 w/o)-titanium and Gd₂O₃ (13.3 w/o)-stainless steel dispersions have been irradiated to approximately 40 percent burnup of the poison atoms. No dimensional changes were noted and the ductility and strength of this material were higher than the boron alloys.^{1e, p. 30}

Moderator, Shielding, and Structural Materials

1. Moderator Materials—*a. Graphite:*

Data obtained from work with graphite irradiated in the Hanford Reactor indicate:^{1b, p. 30}

(1) Increased temperatures tend to decrease the dimensional changes in graphite.

(2) Compressive strength vs. exposure at 30° C. indicates an increase in compressive strength to a maximum ratio of 2.8:1 at 0.5 x 10²⁰ nvt leveling out to an average ratio of 2.3:1 between 3 and 10 x 10²⁰ nvt.

(3) Thermal conductivity vs. exposure at 30° C. shows a decrease with increasing exposure reaching saturation at an exposure of approximately 14 x 10²⁰ nvt.

(4) The electrical resistivity of parallel-cut graphite versus exposure at 30° C. indicates the resistivity increases rapidly with exposure reaching an equilibrium value after approximately 3 x 10²⁰ nvt.

(5) Lattice spacing versus exposure temperature indicates a marked decreased effect in changes in lattice spacing as irradiation temperature is increased from 30° C. to 180° C.

2. Shielding Materials—*a. Lithium Hydride:*

Considerable interest has been taken in LiH as a neutron shield.² Because of this, several cold-pressed compacts of LiH have been irradiated for 10-14 days in thermal fluxes of 10⁹ to 10¹² n/cm² sec at temperatures from 75° F. to 610° F. Some gas evolution took place and the specimens suffered considerable disintegration and growth making them unsuitable unless confined in metallic cans.

b. Magnesium Oxychloride Cement:

MO Cement and MO Cement+20 percent Fe were irradiated to 6.3 x 10¹⁷ nvt and as high as 4.4 x 10¹⁸ nvt. In both types the compressive strength decreased materially.^{1a, p. 29}

MO Cement—7,500 psi (average unirradiated) to 2400 psi (maximum irradiation).

MO Cement+20 percent Fe—9,500 psi (average unirradiated) to 5,000 psi (maximum irradiation).

3. Structural Materials—*a. General:*

A review of the effects of irradiation greater than 10¹⁹ nvt on structural materials is summarized below.^{1a, p. 29}

(1) *Hardness*—usually increases in metals, the greatest increase being in metal in the annealed condition.

(2) *Yield strength*—increased by as much as a factor of 3 in stainless steel and a factor of 2 in annealed carbon steels. Lesser increases in stronger materials.

(3) *Ultimate strength*—tends to increase only 10-30 percent, thus making the yield strength approach the ultimate strength.

(4) *Elongation*—generally decreases. Usually the greater the initial ductility the greater the reduction in elongation.

(5) *Notch-impact*—Carbon steels (irradiated below 200° F.) show an increased fracture transition temperature first observed at integrated fast fluxes as low as 5 x 10¹⁷ nvt. Other steels require 10 or more times this dose to show the same effect.

Austenitic stainless steels, in the subsize Izod test, do not fracture at -320° F. without irradiation. After irradiation, specimens

fracture half-way through at the same test temperature. Other stainless steels such as Type 446 are more affected by irradiation.

(6) Creep—At present, tests involving creep are inconclusive.

b. Beryllium:

Beryllium tubing having wall thicknesses of .060 to 0.100, 4 inch O. D., and 5 feet long as well as tubing 0.12 to 0.25 inch thick, 4 inches O. D., and 6-8 feet long have been successfully extruded.⁷

Some characteristics of the beryllium at a temperature of 1,200° F. are:

Ultimate tensile strength	$15-17 \times 10^3$ psi
Proportional limit	8×10^3 psi
Yield strength	14×10^3 psi

Long term creep tests indicate the limiting stress value of QMV beryllium at a life of 2,500 hours would be: 1,950 psi at 1,200° F. and 700 psi at 1,350° F.

Preliminary tests of alloying iron with the QMV Be have resulted in increasing the tensile strength of the material to 50,000 psi at 1,110° F. and 15,000 psi at 1,475° F.

The corrosion of beryllium was investigated in dry and wet carbon dioxide at atmospheric pressure for the temperature range 500-700° C.

The results of this investigation indicate:⁸

(1) Corrosion rate is negligible up to 500° C. in both dry and wet gas. In this range the weight gain follows the parabolic law $w^2 = kt$ where $k_{\text{dry gas}} \sim 5 \times 10^{-16} \text{ g}^2/\text{cm}^4/\text{sec}^{-1}$ and $k_{\text{wet gas}} \sim 7 \times 10^{-16} \text{ g}^2/\text{cm}^{-4}/\text{sec}^{-1}$.

(2) At 600° C. a product film is formed which is protective in the dry gas, but not in the wet.

(3) At greater than 650° C. the film is no longer protective even to dry CO₂ gas.

c. Magnesium Alloys—Magnox^{3, 4, 5, 6}:

As part of the Calder Hall development program the Magnesium Elektron Ltd. produced a series of Mg alloys containing Be, Ca and Al. These alloys were highly resistant to oxidation and one alloy, Magnox-E, showed no oxidation failure after 2,500 hours at 615° C. in wet CO₂. On melting, these alloys are further protected

against combustion by formation of a protective sheath. Sixty alloys were produced, two typical alloys having the following composition:

Magnox-E 1 percent Al, 0.05 percent Be, 0.1 percent Ca in 99.95 percent pure Mg.

Magnox-C 0.7-0.9 percent Al, 0.005 percent Be in 99.95 percent pure Mg.

Further tests indicated Magnox alloys were prone to weld cracking during can fabrication and it was discovered that Ca should be kept below .05 percent or omitted, and Al should not exceed 10 percent. Corrosion tests in CO₂ with varying amounts of moisture indicated that Magnox-C was their best alloy. Magnox-C contains no Ca and tests run in dry CO₂ at 8 atmospheres pressure gave an oxidation rate of 0.1 mg/cm²/5 mo. Table VII-1 lists some of the oxidation rates of various materials.

Ignition tests on Magnox alloys revealed an ignition temperature of approximately 640° C. at 11 atmospheres pressures of CO₂ for all alloys listed. No ignition occurred at temperatures of up to 700° C. with CO₂ at atmospheric pressure and at a flow rate of 20 ft³/hr.

The Magnox alloys are extremely sensitive to certain metallic contaminants. Table VII-2 is a list of the contaminants and the temperature at which complete penetration can be expected.

Mg alloys do not form any intermetallic compounds with U metal and consequently no barrier between U and Mg is required.

Data on the tensile stress and elongation at elevated temperatures for Magnox-C appears in table VII-3.

Table VII-4 indicates the creep strain vs. time of Magnox-C at 350° C.

The South of Scotland Electricity Board Station (G. E. C.-Simon Carves Design) has specified a new Magnox alloy called Magnox A-12 for their fuel canning material. The operating can temperature will be approximately 797° F. and the firm's metallurgical research work has shown this material to be satisfactory up to approximately 900° F. As yet no detailed information is available on this alloy.

TABLE VII-1*-OXIDATION OF MATERIALS IN CO₂(Test carried out at a temperature of 400° C. in dry CO₂ at 8 atmospheres pressure.)

Material	Oxidation
18/8/1 stainless steel	No measurable attack
Commercial Al	after 5 months.
Sintered Al powder	
Pure Mg	
Magnox-C	0.1 mg/cm ² /5 mo.
Mg alloy AM503	
Al alloy RR58	
Al alloy RR57	
Mg alloy ZT1	
Mg alloy ZREI	
Magnox-E	
Alloy steel (Jessops G1)	0.15 to 1.11 mg/cm ² /5 mo.
Mild steel (0.05% C)	
Sintered Mg powder	
Pressure-vessel steel	
Heat-exchanger steel	
Cast iron	2.72 mg/cm ² /5 mo.

*The effect of moisture (0.15 wt. %) was to increase the rate of oxidation by a factor falling from 4 in the first month to 2 after 2 months.

TABLE VII-2—PENETRATION OF MAGNOX BY CONTAMINANTS

Contaminant	Temperature for complete can penetration
Mercury	All temperatures.
Tin	Above 200° C.
Lead, bismuth	Above 250° C.
Cadmium, zinc	Above 300° C.
Aluminum, calcium	Above 400° C.
Silver, copper	Above 450° C.
Nickel	Above 500° C.

TABLE VII-3—ULTIMATE TENSILE STRESS AND ELONGATION OF MAGNOX C VS. TEMPERATURE

(Strain rate 18 percent/hr.)

Temperature ° C.	Ultimate tensile stress # /in ²	Elongation (percent)
° F.		
50	122	16,000
100	212	12,800
150	302	9,200
200	392	6,200
250	482	4,200
300	572	3,000
350	662	1,600
400	752	1,000

TABLE VII-4—CREEP STRAIN VS. TIME

(Magnox C at 350° C.)

Creep strain (percent)	Time (hours)						
	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0
1							400
2	10	20	40	60	60	200	1,100
5	14	30	75	160	190	500	-----
10	18	60	140	280	390	1,050	-----
15	25	80	185	390	585	-----	-----
20		100		480	780	-----	-----
22			*220				-----
25						*2,200	-----
26		*115					-----
30				*580			-----
36		*35					-----
52						*1,460	-----
	800	700	600	500	400	300	200
							Stress # /sq. in.

*Extrapolated to rupture.

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PROGRESS ON SPECIFIC REACTOR TYPES

GAS-COOLED REACTORS

Calder Hall and Related Types

The largest single program for the development of power reactors—the British gas-cooled, graphite-moderated, natural uranium program—achieved a major goal in putting the first reactor, Calder Hall, into operation in October 1956. This reactor represents the successful solution of a great number of technical problems and lays the groundwork for the future reactors. The value of this work is already evident in the improvement in the reactors now being built or in the advanced planning stage. It is worthy of note that the already ambitious British program is being accelerated further on the basis of the success of the first reactor system.

While a myriad of problems always present themselves in a large reactor, certain of these are of crucial importance and solutions must be found. Among these for the Calder Hall type reactors were the development of materials of low absorption for use in CO₂ at high temperature, a design of fuel slug to achieve the necessary heat transfer rates, and the fabrication of pressure vessels of sufficient size to contain the reactor. The solution of these problems and the continued improvement has led to the design of the higher performance reactors. The jacket material is discussed under "Other Reactor Materials," (pp. 30-33). A discussion of the fin arrangement and heat transfer is found on pp. 5-10. The techniques for fabricating large vessels in the field have been developed for metal up to two inches in thickness and are now being applied to thicknesses of three inches for the new reactor vessels. Equipment for handling the massive pieces, for field welding, and for annealing and

stress relieving has been worked out and applied.

Table VIII-1 lists the principal characteristics of the Calder Hall reactor. References 1 through 5 present the design of Calder Hall and the other British gas-cooled reactors in considerable detail.

The British report complete satisfaction after eight months of operation.⁵ Most of the difficulties which showed up in the early stages of operation have been cleared up. The creep rate of the lower elements under load is greater than anticipated, requiring additional lateral support for the fuel element. The improved element is used in No. 2 reactor and will be used to replace spent slugs in No. 1 at the next reloading.

The CO₂ leakage is being remedied by conventional methods. Meanwhile, no difficulty is encountered due to radioactivity of the gas.

The operation of the station is said to be smooth and flexible with a reduction of power output from 100 percent to 5 or 10 percent and back to 100 percent being possible within a half hour.

The new British industrial design proposals^{1, 3, 5} are all based on the Calder Hall plant as a prototype, but differ substantially from it in design details, with a large advance in power output.

A notable advance in design is the increase in size and heat output of the reactor core. All of the designs propose a heat output of more than 500 MW, mainly due to the increase in the core size. The extent of the advance in power output must be modified somewhat by the fact that the Calder Hall design was optimized for plutonium, rather than for power production.

Another significant advance in the new designs is the introduction of fuel changing under load. Designs have been developed for fuel handling machinery which will enable loading and unloading of fuel elements without interrupting the operation of the plant.

TABLE VIII-1

<i>Overall:</i>	
Reactor thermal output MW.....	182
Reactor coolant outlet temperature, °F.....	637
Reactor coolant inlet temperature, °F.....	284
Reactor coolant mass flow, lb/sec.....	1,964
Net electrical output, MW.....	39
Net overall thermal efficiency, percent.....	21.5
<i>Reactor core:</i>	
Fuel.....	Natural U
Total weight of fuel, tons.....	130
Number of fuel-element channels.....	1,696
	[256 are 4.16 in.
Diameters of fuel-element channels.....	576 are 3.95 in.
	864 are 3.61 in.
Pitch of channels, in.....	8
Number of control-rod channels.....	112
Diameter of control-rod channels, in.....	3.25
Nominal core height, ft.....	21
Nominal core diameter, ft.....	31
Height of graphite, ft.....	27
Diameter of graphite (across corners), ft.....	36
Machined weight of graphite, tons.....	1,146
Core graphite density, gm/cm³.....	1.73
Core graphite absorption cross section (measured in air in GLEEP), millibarns.....	4.0
Reflector graphite density, gm/cm³.....	1.60
Reflector graphite absorption cross section (measured in air in GLEEP), millibarns.....	4.8
U rod diameter, in.....	1.15
Length of U in channel, ft.....	20
Density of U, gm/cm³.....	18.7
Absorption cross section of cans and supports, em²/ft U.....	0.51
<i>Reactor Pressure Vessel:</i>	
Inside diameter, ft.....	37
Thickness of shell, in.....	2.0
Design stress at 752° F., psi.....	12,000
Working pressure, psig.....	100
<i>Main CO₂ Ductwork:</i>	
Numbers of ducts in parallel.....	4
Diameter of ducts, ft.....	4.5
Mean coolant velocity (hot ducts), ft/sec.....	74
Mean coolant velocity (cool ducts), ft/sec.....	49
<i>Main CO₂ Circulators:</i>	
Number of circulators.....	4
CO ₂ flow rate per circulator lb/sec.....	500

TABLE VIII-1--Continued

<i>Main CO₂ Circulators—Continued</i>	
Pressure rise in circulator, psi.....	5.53
Isentropic efficiency at design point, percent.....	78
Shaft power to circulator motor, h. p.....	1,495
Electrical power required (4 circulators), MW.....	5.44

Main Heat Exchangers:

Number of heat exchangers.....	4
H. P. steam flow rate, lb/hr.....	396,000
H. P. steam exit temperature, °F.....	595
H. P. steam exit pressure, psia.....	210
L. P. steam flow rate, lb/hr.....	118,600
L. P. steam exit temperature.....	350° F.
L. P. steam exit pressure, psia.....	63
Feedwater temperature, °F.....	100

Turbo-Alternators:

Number of sets per reactor.....	2
H. P. steam temperature, °F.....	at 590
H. P. steam pressure, psia.....	T. 200
L. P. steam temperature, °F.....	S. 340
L. P. steam pressure, psia.....	V. 53
Flow-rate to condenser per set, lb/hr.....	257,300
Vacuum at turbine exhaust, in. Hg.....	28.25
Gross electrical output of each turbo-alternator, MW.....	21.0

*Russian Gas-Cooled Heavy-Water Reactor*⁶

This reactor proposal is of interest because it represents viewpoints quite different from those current in this country. For example, it is stated that gas cooling permits a reduction of the quantity of heavy water required per megawatt of electrical power by a factor of 3 or 4 below the quantity required when cooling by heavy water is employed. Another interesting conclusion is that the optimum steam pressure for use with a D₂O-cooled reactor having zirconium-clad fuel elements with a temperature limit 300° C. is approximately 5 atmospheres. This "optimum" is the pressure at which the power output per ton of D₂O is maximized.

The gas-cooled reactor is a calandria type with a gas inlet temperature of 90° C. and an outlet temperature of 420° C. The fuel elements are thin rods or plates of natural uranium, clad with an unspecified light metal which will withstand a maximum temperature of 550° C. The coolant is CO₂ at 40 to 60

atmospheres, and steam is to be generated at 400° C. and at two pressures: 29 atmospheres and 2 atmospheres. The quite low specific D₂O inventory of 0.2 to 0.4 ton per megawatt of electrical power is claimed. It appears that this low inventory is achieved at the price of expensive reactor design and probably some loss of reactivity lifetime. It would appear to indicate that D₂O is quite expensive in the Soviet Union. The report states that it is proposed to commission in the Soviet Union an atomic power station of this kind with an electrical output of 100 to 200 megawatts.

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BOILING WATER REACTORS— THE EXPERIMENTAL BOILING WATER REACTOR (EBWR)*

Assembly of reactor internals and initial loading of fuel were carried out in the EBWR during the latter part of November 1956. Criticality was achieved December 1 and full

*This information was obtained from the technical briefing session conducted by ANL in connection with the February 9 dedication ceremonies plus subsequent discussions with EBWR project personnel.

After this was written the Atomic Energy Commission published a detailed description of this reactor. The book entitled "The EBWR" is available from the Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C., for \$2.25 a copy (paperbound).

power output of 5,000 electrical kilowatts was reached on December 29. During January 1957 the reactor and power plant equipment were run on an intermittent basis while overall performance was being checked. On February 9 the plant was formally dedicated and placed on a more or less continuous basis.

The specification of 1,000 ft³/day maximum leakage from the EBWR containment building under a pressure differential of 15 psig was satisfied prior to startup but not without some difficulty, particularly in the seals for the multitudinous electric cables passing from the power plant through the steel shell to the control room. The leakage was measured while holding the building under 15 psig internal pressure for three days. An earlier test on the steel shell prior to making openings for airlocks, air ducts, pipes, conduits, cables, etc., revealed no leaks at all.

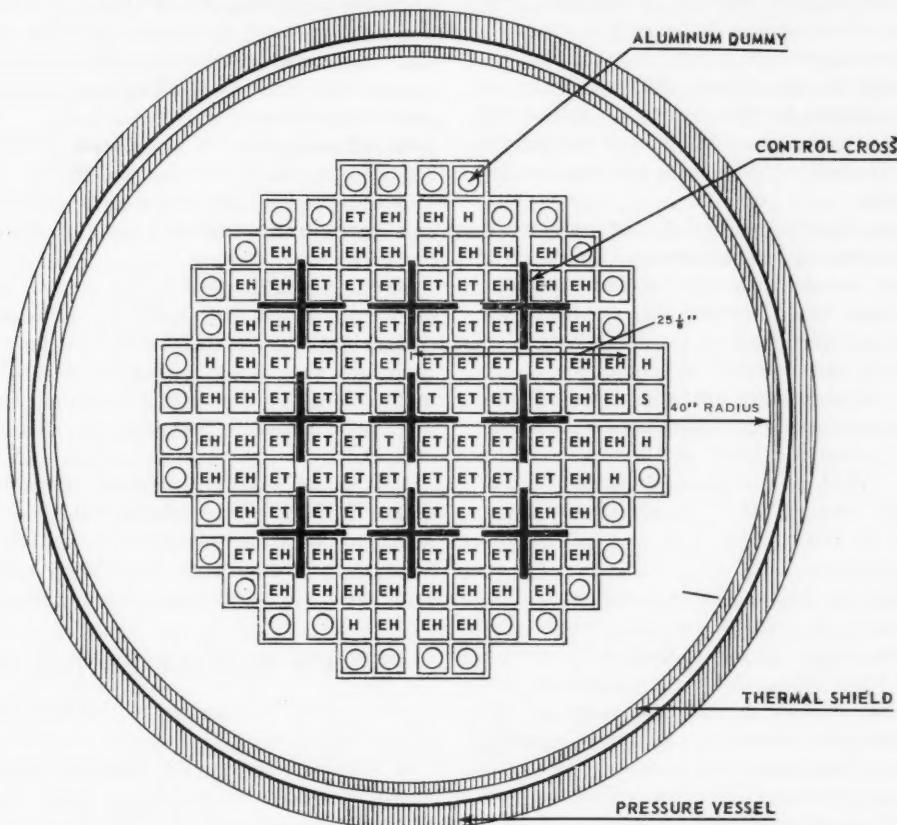
In lieu of the usual expensive and time consuming series of critical experiments prior to finalization of fuel loading, the Argonne National Laboratory decided to fabricate a variety of fuel elements to cover all uncertainties in physics characteristics of the EBWR and to determine the best arrangement of this fuel in pre-startup experiments within the reactor itself. Major uncertainties were recognized in the U²³⁵ content required for operation at rated power level in a 4-foot diameter core and in the water to uranium volume ratio required to give a very slightly negative steam void coefficient of reactivity at room temperature. Accordingly fuel elements were provided having both natural and double natural (1.44 percent) U²³⁵ content. For each enrichment, fuel assemblies having two plate thicknesses were provided. These four types of fuel assemblies covered an uncertainty of approximately 7 percent in reactivity and allowed the water to uranium volume ratio to be varied between 2.8 and 4.4.

The initial fuel loading experiments in the EBWR showed the reactivity to be less than expected. Criticality was reached with 81 fuel assemblies as compared with a prior estimate of 42. This increase in critical size corresponds

to about 3.4 percent in reactivity, which is approximately the same as the probable error calculated in a previous analysis of uncertainties in constants and theoretical methods. This 3.4 percent discrepancy was covered twice over by the variety of fuel provided, together with a possible increase in reactor diameter to 5 feet.

The pre-startup experiments on EBWR in

which styroform was used to simulate steam showed a higher steam void coefficient of reactivity than calculated for the case of equal numbers of thick and thin plate fuel assemblies. This was corrected by shifting the thick and thin plate fuel assemblies provided for that purpose, to increase the water to metal ratio in the center of the reactor.



LEGEND:

- EH. Thick Plate Enriched (1.44% U²³⁵) Assemblies
- ET. Thin Plate Enriched (1.44% U²³⁵) Assemblies
- H. Thick Plate Natural Uranium Assemblies
- T. Thin Plate Natural Uranium Assemblies
- S. Neutron Source

Figure 6. Fuel loading pattern in EBWR.

The final loading of the EBWR core is illustrated in figure 6. Thin plate fuel assemblies (water to uranium volume ratio 4.4) are concentrated in the central zone and thick plate assemblies (water to uranium volume ratio 2.8) in the outer zone. Except for two natural uranium assemblies located at the center to flatten the flux distribution, the central zone consists entirely of enriched assemblies. Six other natural assemblies are located in the outer region for adjustment of reactivity to the desired value in the 4-foot diameter core. A strong antimony beryllium neutron source (S) is kept in the reactor at all times to supply the neutron flux needed for reliable instrumentation of startups.

Analysis of physics data obtained prior to and during startup is still underway. Tentatively it appears that the reactivity loss in heating the reactor from room temperature to operating temperature (448° F.) is in the neighborhood of 3 percent k, and the loss due to operating steam voids at 20 megawatts heat output is between 1 percent and 2 percent depending on the variation in control rod bank position with xenon content. With equilibrium xenon and samarium, the present core has about 1 percent excess k to cover long term irradiation and shutdown xenon.

Stability of the reactor at 600 psig and 20 megawatts heat output has been remarkably good, even when the upper half of the core is nullified by insertion of the entire bank of nine control rods beyond the half-way mark to hold down reactivity when no xenon is present. Response of the reactor to imposed changes in ambient pressure and in feedwater flow has been essentially as expected.

Generally speaking, operation of the entire powerplant has, thus far, been highly satisfactory. Radioactivity levels in powerplant equipment have been lower than expected. Typical values near major equipment during full power operation are shown in table IX-1.

The most disconcerting observation in connection with radioactivity has been the confirmed presence of fission product gases in the exhaust from the main condenser air ejectors

TABLE IX-1—GAMMA ACTIVITY IN EBWR PLANT

Location	mr./hr.
Main floor area-----	1 to 2
Outside turbine casing-----	8 to 10
Outside steam dryer-----	130 to 150
Outside air ejector aftercoolers-----	300 to 450
Outside condenser hotwell-----	30 to 40

since the beginning of operation. A quantitative evaluation of the fission products showed that they could result from surface contamination of fuel elements totalling only a few milligrams of uranium. This fact, together with the constant magnitude of the fission product activity, has ruled out the possibility of fuel cladding failure. Attempts to trap the xenon and krypton on cold activated charcoal are underway.

The experience with fission gas in the air discharged from the EBWR condenser has shown that monitoring of these discharge gases from the condenser is indeed a very sensitive method for detecting fuel cladding failures. This is a standard procedure in the EBWR system.

It is understood that a report describing the design of the EBWR in detail will be issued by ANL shortly (see p. 36) and that an information meeting for industry in which early operating experience will be discussed is being planned.

HEAVY WATER MODERATED REACTORS

A study of reactivity lifetime by W. B. Lewis,¹ leads to a conclusion that the possibility exists of attaining reactivity lifetime of approximately 10,000 mwd/ton with natural uranium fuel. The conditions to be met for such performance are that the fuel be used at uniformly-graded irradiation, and that the neutron losses be low. Lewis estimates that the attainment of sufficiently low neutron losses in a graphite moderated reactor is improbable, but that their attainment in a D₂O moderated reactor with a coolant of low absorption (D₂O or gas) may be feasible.

Neutron losses are expressed in the unit of equivalent barns per U^{235} atom. The analysis indicates that the losses must be held to about 110 barns per U^{235} atom or less if the lifetime of 10,000 mwd/ton is to be reached. The analysis takes into account separately all losses to fission products, including xenon and samarium, and hence the specified loss of 110 barns per U^{235} atom refers only to losses by neutron leakage and parasitic absorption by moderator, structural materials, and control elements. As an example Lewis postulates that the losses might be broken down in a typical reactor as 25 barns per U^{235} atom for all parasitic absorption and 85 barns per U^{235} atom for leakage. The parasitic absorption specification means simply that the total macroscopic cross section of all parasitic absorbers in the reactor, divided by the total number of U^{235} atoms in the reactor, would give the quotient of 25 barns per U^{235} atom. In this computation the actual absorption cross section must be increased by the disadvantage factor where applicable, to give the effective cross section. As a numerical example, if all of the parasitic absorption were by zirconium, and if the disadvantage factor were unity, a neutron loss of 25 barns per U^{235} atom would result if approximately equal volumes of zirconium and natural uranium metal were contained in the reactor.

In order to specify the leakage in terms of barns per U^{235} atom, the neutron loss by leakage is represented as an absorption loss by a fictitious absorber. The relation between this specification of leakage and the more usual reactor characteristics is approximately:

$$\text{Leakage in barns}/U^{235} \text{ atom} = \eta_{25} \epsilon \sigma_{25} \frac{M^2 B^2}{1 + M^2 B^2}$$

Where η_{25} is the number of neutrons liberated per neutron absorbed by U^{235} , ϵ is the fast fission factor, σ_{25} is the absorption cross section of U^{235} (in barns), M^2 is the migration area and B^2 is the geometric buckling of the reactor. A value of 85 barns per U^{235} atom corresponds approximately to $M^2 B^2 = 0.07$.

The meaning of uniformly-graded irradiation is that the reactor fuel consists of a uniform

mixture of fuel having all degrees of exposure, from 0 to the maximum, and that the amount of fuel in unit exposure interval is constant over the exposure range. With this type of loading, that part of the fuel which is in the later stages of irradiation may be incapable of supplying enough neutrons per neutron absorbed to maintain a chain reaction, but the neutrons necessary for maintaining the reaction in the reactor as a whole are supplied by the excess neutrons from the newer portion of the fuel. Lewis demonstrates that in this case maximum lifetime is attained by keeping the resonance escape probability high and by using a large fraction of the excess neutrons, not for conversion, but for supplying neutrons to the older portions of the fuel charge. He estimates that a practical reactor might operate with a resonance escape probability of 0.90 and a conversion ratio of approximately 0.78. It should be emphasized that a reactor with the postulated low neutron losses could have a conversion ratio as high as approximately 0.90 if the resonance escape probability were decreased to give maximum conversion ratio.

Although the purpose of the paper is to develop the relationships which have a bearing on reactivity lifetime, and no reactor design is attempted, some suggestions are made as to reactor design features which would make possible an approach to the conditions established for long burnup. It is suggested that a pressurized-tube type reactor, employing D_2O or gas as the coolant, would make it possible to approximate closely the condition of uniformly-graded irradiation. This condition would of course exist only after the reactor had operated long enough to approach its equilibrium loading cycle. It is suggested that reloading be done at intervals of approximately 5 percent of the total irradiation lifetime of the fuel, approximately 5 percent of the fuel of course being replaced at each reloading. If the total irradiation lifetime is 10,000 mwd/ton this would mean that reloading would occur at intervals of 500 mwd/ton. It is evident that such frequent reloading would not be desirable with a pressure vessel type reactor. Lewis suggests

that reloading might be possible during operation. He postulates also that reloading be possible from either end of the reactor in order that spent fuel in any given fuel channel be adjacent to fresh fuel in other channels.

It may be anticipated that the construction of a pressurized tube reactor with low parasitic neutron losses postulated by Lewis will not be easy, and in fact cannot be presumed to be possible until detailed design studies have been made. Nevertheless, even if it should prove necessary to accept somewhat higher neutron losses, the general scheme of reactor design and operation proposed appears to be a most attractive one, and the potentialities of the natural uranium-heavy water reactor for low fuel cost are of great importance.

Reference

1. W. B. Lewis, Low Cost Fueling Without Recycling, AECL-382, December, 1956. (Unclassified Canadian report.)

TRAINING REACTORS— ARGONAUT^{1,2}

Argonaut is a thermal reactor composed of an annular cylindrical core moderated by water and graphite, and reflected internally and externally by graphite. It is designed for use as a training reactor and for conducting experiments in reactor physics.

The core is heterogeneous; composed of aluminum-uranium alloy fuel plates clad with aluminum. A longer neutron lifetime than that characteristic of H₂O-reflected, H₂O-moderated systems is anticipated for reasons of the annular core geometry and graphite reflectors. The calculated lifetime is 1.9×10^4 sec.

Interstitial water between fuel elements, which presumably is the cause for positive void coefficients sometimes observed in similar reactors, has been avoided by use of the graphite separators.

The reactor is capable of operation at a maximum of 10 kw of thermal power, at which level only brief tests are recommended.

The large negative void coefficient means that a rather large increase in reactivity can be removed by voids introduced through boiling. The reactor would finally shut itself down when enough water had been boiled away. The negative temperature coefficient combined with a low water circulation rate and cooling capability is sufficient to prevent a slow, steady increase in power level. The same considerations limit the use of the reactor at high power, since inevitably there will be a gradual temperature increase at such a power.

It has been observed that a void space between fuel boxes gave a greater reactivity than the same space filled with water. It was also observed in experiments that the sign of the uniform void coefficient, even without the graphite spacers, is negative.

When water is admitted to the fuel region the graphite filler pieces become submerged and hence must be waterproofed. As an inexpensive substitute for aluminum cladding, an aluminum-Krylon plastic spray coat is used. Irradiation in the CP-5 reactor comparable to several years of operation of Argonaut caused no degradation of the coating.

A fabrication technique for making fuel plates containing 35 wt percent of 20 percent enriched U₃O₈ was developed. A hot extrusion of a mixture of U₃O₈ and 2S aluminum powder gave plates with negligible void volume. Aluminum powder and U₃O₈ in the proper ratio were placed in a 3½-inch diameter vented aluminum can, heated to 483° C. sealed and then extruded in a 400 ton horizontal press. The resulting fuel sheet, approximately 17 feet long, was cut into sections two feet long. Over-all dimensions of the plates were 24 inches long, 2.84 inches wide and 0.098 inch thick.

A clad averaging two mils thick covered the plate except on the ends at the point of the cut off and at some scratch points along the surface. Exposed portions of the fuel matrix present no corrosion problems; however, a plastic spray is applied to stop fission recoils.

The uranium oxide content of the plates varied; those cut from the ends of the extrusion contain somewhat less U₃O₈ than the average.

The composition of each plate is:

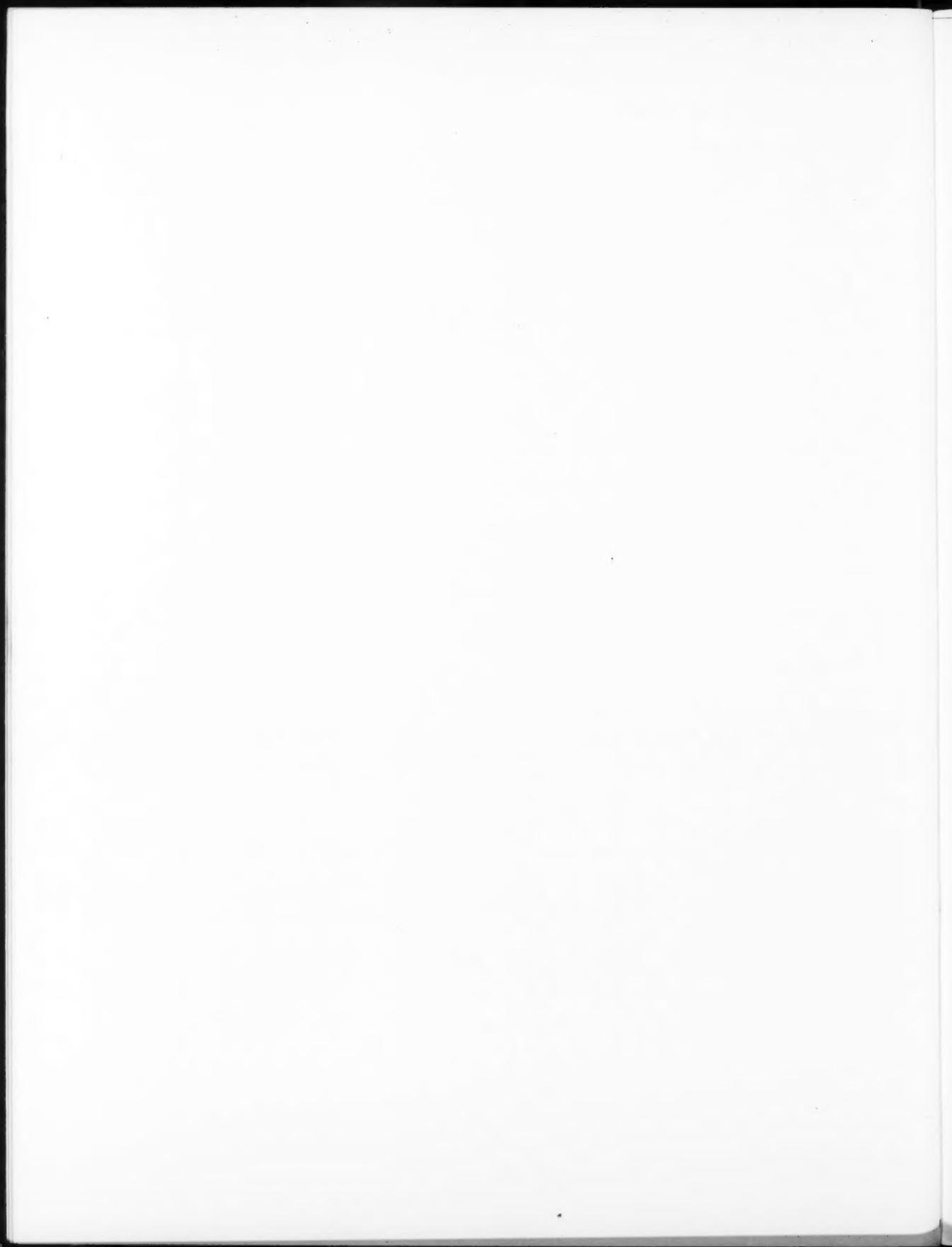
U^{235}	19.6 gm $\pm 10\%$
U_3O_8	114 gm
Al	248 gm

Cooling water for the reactor is limited to prevent continuous operation at 10 kw. After several hours at the maximum power, the mod-

erator temperature will have increased enough to exhaust the excess reactivity.

References

1. D. H. Lennox and B. I. Spinrad, A Generalized Reactor Facility for Nuclear Technology Training and Research: Interim Report on Argonaut, ANL-5552, March 1956. (Unclassified AEC report.)
2. D. H. Lennox and C. N. Kelber, Summary Report on the Hazards of the Argonaut Reactor, ANL-5647, December 1956. (Unclassified AEC report.)



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